



CALIBRATION HANDBOOK:

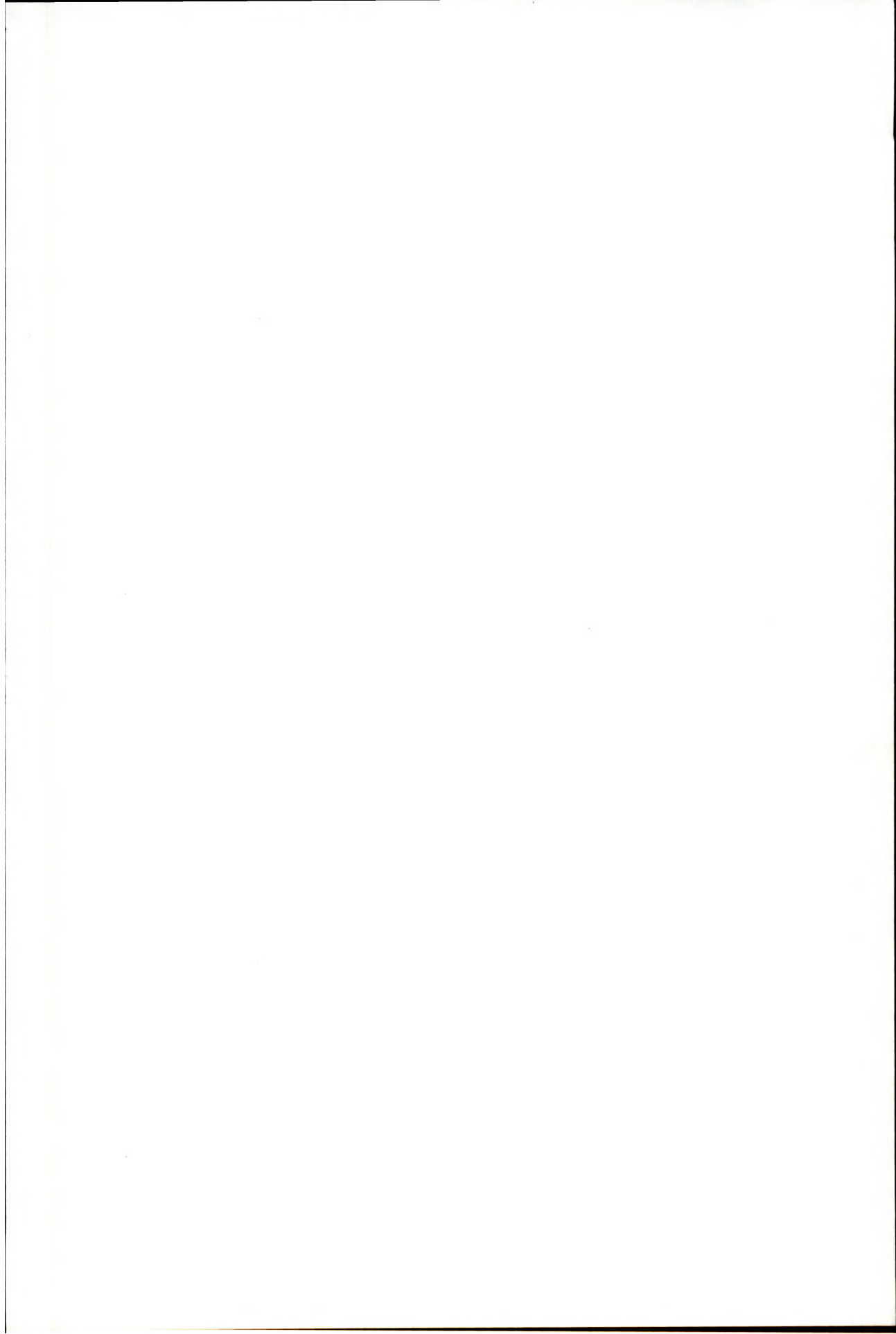
**IONIZING RADIATION
MEASURING INSTRUMENTS**



Edited by

G. Lakos

**Calibration Coordinating Group
Department of Defense
Joint Technical Coordinating Group
for Metrology and Calibration**



CALIBRATION HANDBOOK: IONIZING RADIATION MEASURING INSTRUMENTS

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July 1, 1983

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for Metrology and Calibration**

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PREFACE

This handbook was written to satisfy the need for a single comprehensive source of information on the principles of operation and calibration of all major classes of ionizing radiation measuring instruments. The handbook is a project of the Radiological Working Group of the Department of Defense Joint Technical Coordinating Group for Metrology and Calibration (JTTCG - METCAL) and is addressed primarily to the needs of the RADIAC (Radioactivity Detection Indication and Computation) programs of the Army, Navy, and Air Force. An appendix describing the armed forces radiation protection programs is included to provide the reader with insight on how the principles of instrument operation and calibration are implemented for radiation protection purposes.

The handbook is intended to be of special interest to management and supervisory personnel of RADIAC programs at all hierarchical levels. It is not meant to be a rigorous treatise for use by instrument designers or for specialists engaged in sophisticated measurements. A number of excellent texts are available for that purpose. The emphasis is on basic coverage of all areas necessary to a complete understanding of the principles involved in the operation and calibration of ionizing radiation measuring instruments. The handbook includes a chapter on nuclear radiation fundamentals which can serve as a short refresher course for the reader not previously active in the field (Chapter 1), an overview of the principles and practices underlying instrument calibration (Chapter 2), nine chapters covering the major classes of radiation detection instruments, and a chapter on radiation biology and radiation protection to acquaint the reader with the effects of ionizing radiation on biological systems and with the measures being taken to protect operating personnel (Chapter 12). An extensive list of references is provided for further study.

The SI system of units has been used throughout the handbook with the older traditional units sometimes given in parenthesis. Readers not familiar with the newer SI units will find the definitions reviewed in Chapter 1.

Funding for the handbook was provided by the Army, Navy, and Air Force through the DOD/CCG Radiological Working Group of the JTTCG-METCAL. At the inception of this project the radiological working group was composed of Mr. Charles Harper, Army, Redstone Arsenal, Alabama, Mr. Charles Hollander (Chairman), Navy, NAVELEX, Washington, D.C., and Mr. Charles Price, Air Force, AGMC, Newark AFS, Ohio, with project management provided by Dr. P.W. Hesse, NSWC/WO, Maryland. More recently, the Navy's representation on the working group has been assumed by Mr. Michael Schaeffer (NAVELEX, Wash., D.C.), and the Air Force slot has been filled by Mr. Robert Burns (Newark AFS, Ohio). The bulk of the handbook was prepared, under contract, by RTS Laboratories, Inc., Gainesville, Florida.

Many individuals have contributed to the successful completion of this project. Aside from the handbook sponsors and the authors of the individual chapters, special acknowledgement is made of the invaluable contributions made by the reviewers. A partial list includes R.S. Allgaier, NSWC/WO, Maryland, J.W. Duley, NAVSEASYS COM, Wash., D.C., J.H. Schulman, Bethesda, Maryland, and staff members of the National Bureau of Standards. NBS reviewers include Messrs. R.L. Ayres, S.R. Domen, E.H. Eisenhower, R.F. Fleming, H.T. Heaton II, T.P. Loftus, E.D. McGarry, W.L. McLaughlin, J.S. O'Connell, R.B. Schwartz, L.A. Slaback, and C.G. Soares.

H.T. Heaton II deserves special citation for his very able, competent and professional contributions to this project. His efforts, together with those of E.H. Eisenhower, were clearly "beyond the call of duty" and are gratefully acknowledged and deeply appreciated.

A debt of gratitude is owed the authors and publishers who so kindly gave their permission to reproduce figures and tables from copyrighted material. Individual credit is given in captions and by citing a reference to the original source.

Finally, Karen Bryant is commended for her dedicated work in typing the many drafts of the manuscript and for her expert assistance in many other areas in the production of this handbook.

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Editor

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CHAPTER 1

FUNDAMENTALS OF NUCLEAR RADIATION SCIENCE

R. T. SCHNEIDER and J. D. COX

1.1 INTRODUCTION

The detection and measurement of ionizing radiation is the basis of nuclear science and technology. Applications abound in such diverse fields as nuclear power, high-energy physics, non-destructive testing, and medicine. In the context of this handbook ionizing radiation is interpreted to mean any electromagnetic or particulate radiation capable of producing ions, directly or indirectly, in its passage through matter. This includes alpha particles, beta particles, gamma rays, x rays, neutrons, high-speed electrons, high-speed protons and other particles capable of producing ions. The term radiation source is used in the broad sense, i.e., any material, device, or equipment which generates or is capable of generating ionizing radiation. This includes naturally-occurring and artificially-induced radioactive materials; nuclear reactors, particle generators, and accelerators; medical, dental, and industrial radiographic and fluoroscopic equipment; electromagnetic wave generators capable of producing ionizing radiation; and contaminated material. Because of the important role laboratory-scale sources of ionizing radiation play in either the testing or calibration of radiation detectors, a brief review of common sources of ionizing radiation is presented in this chapter.

Detection and measurement of ionizing radiation is normally accomplished by collecting the electron-ion pairs (or electron-hole pairs in the case of solid-state detectors) by the use of appropriate electrical circuits. The study of ionizing radiation, therefore, requires a basic understanding of the physical processes involved in the interaction of radiation with matter leading to the creation of electron-ion (or electron-hole) pairs. These processes involve the transfer of energy from the emitting source to the detector medium where they interact to an extent determined by the characteristics of the radiation and the nature of the detector material. Strong interaction between the radiation (e.g., alpha particles) and the detector medium results in rapid energy transfer and strong attenuation of the incident radiation. Weak interaction (e.g., with neutrons), on the other hand, results in the ionizing radiation having appreciable range, and necessitating more elaborate shielding measures. Clearly, all detection methods require an efficient

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interaction mechanism to maximize the detector signal required for accurate detection and measurement of the incident radiation. For this reason, considerable attention is devoted in this chapter to a review of the fundamentals of the interaction of ionizing radiation with matter.

Ionizing radiation, unlike visible light and infrared radiation, cannot be detected by the common human senses. As a result, detectors must be used that provide a conversion of the incident particle or photon to a readily measured phenomenon. Fortunately, a number of physical principles can be used to design suitable detector systems. Ideally, each detector system should respond principally to only one type of radiation, although in many cases, high levels of different radiation types can cause troublesome interference. As a result, a large variety of detector devices and detector systems have been developed for different applications. These systems can be conveniently grouped into classes of instruments with each class based on a common principle of operation. Subsequent chapters of this handbook describe the principles of operation, operating characteristics, and typical applications of the major classes of detector systems.

The chapter is concluded with a brief treatment of statistics and errors frequently associated with radiation measurements. The reader is reminded that the material in this chapter is meant only as a review of the physical processes that underlie the design, operation, and calibration of instruments used to detect and measure ionizing radiation. A list of references is provided for the reader requiring a more in-depth treatment of the subject before proceeding with subsequent chapters of the handbook.

1.2 RADIATION SOURCES

1.2.1 Introduction

There are many sources of ionizing radiation. They can be characterized in terms of how they function and the types and intensity of radiation they produce.

The most common source of radiation is background radiation. Background radiation is produced from naturally occurring radioisotopes found in the earth, from cosmic radiation, and even man made sources. Background radiation is principally gamma radiation and has an intensity typically corresponding to a few $\mu\text{R/hr}$. The integrated dose over a year's time usually amounts to about 100 millirem but will vary depending on location, building materials, and altitude.

Of the man made radiation sources, the most common is the isotopic source, intentionally produced by irradiating material with beams from accelerators or as byproducts of the nuclear fuel cycle.

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These sources consist of a single radioisotope that may produce one or more decay products. They can be used as calibration and irradiation sources and have activities ranging from microcuries to thousands of curies. One can make a neutron source by combining certain radioactive isotopes with certain low atomic number isotopes. An example of this type of source is the plutonium-beryllium source. This source is made from a mixture of plutonium-239 and beryllium-9. First, the plutonium decays by alpha emission. This alpha particle interacts with a beryllium-9 nucleus creating a carbon-12 nucleus and a neutron (and a 4 MeV gamma ray). Nuclear reactions of this type can be abbreviated as follows: ${}^9\text{Be}(\alpha, n){}^{12}\text{C}$. Where ${}^9\text{Be}$ is the original nucleus, α is the bombarding particle, ${}^{12}\text{C}$ is the product nucleus, and n is the principal form of radiation generated by the nuclear reaction.

The largest radiation sources are machine sources. They can be classified into two types: accelerators and reactors. The most common type of accelerator source is the x-ray machine. This source works by accelerating electrons to a high velocity and bombarding a solid target. The kinetic energy of the electrons is released in the form of x rays and is called bremsstrahlung. Nuclear reactors are the most intense radiation sources. A typical power reactor may produce radiation fields up to 10^9 R/hr in the core.

1.2.2 X rays and Gamma rays

X rays and gamma rays are high energy photons (typically a thousand to a million times more energetic than visible light photons). They are identical in form and substance. However, gamma rays originate from the nucleus of an atom, whereas x rays result from interactions with the electron shell of an atom. Like all photons, they have no rest mass and cannot be deflected by electric or magnetic fields.

The most common source of gamma rays for laboratory use is from radioisotope sources that first undergo beta decay to a daughter nucleus in an excited state. The gamma ray is then emitted by the daughter in a transition to a lower lying nuclear level. Two of the most common gamma-ray sources are cobalt-60 and cesium-137, although the gamma rays are actually emitted by their respective daughters, nickel-60 and barium-137. In the case of the cobalt-60 source, there is a cascade in the nickel-60 nucleus giving rise to two gamma rays at 1.17 MeV and 1.33 MeV.

Gamma-ray emission from an excited nucleus is a competing process with internal conversion (see Section 1.2.3) which in turn leads to the emission of characteristic x rays (characteristic x rays will be discussed later in this section).

The gamma rays emitted from any one nuclear transition are nearly monoenergetic, thus gamma-ray sources make ideal reference sources for any radiation measuring laboratory. Radioisotope gamma-ray sources are available with a variety of energies up to

about 3 MeV and activities of a few microcuries (10^5 Bq). At this activity level they are intense enough to calibrate detector systems yet not a biological hazard. These sources are usually encapsulated with a material thick enough to stop the primary decay products produced from the decay of the parent nucleus so that only the desired gamma ray emerges from the source. However, in some cases secondary radiation such as bremsstrahlung and annihilation radiation can be significant. If the sources are to be used in accurate efficiency calibrations, their absolute activity must be known. In this case, it is not desirable to have a source with thick encapsulation which introduces attenuation and scattering. Therefore, radiation sources are available with much thinner encapsulation. However, the other forms of radiation accompanying the desired gamma radiation must be dealt with.

A practical limit for gamma-ray energies with beta decay radioisotopes is about 3 MeV. Radioisotope sources with higher energy gamma rays have such short half-lives that they are not convenient for many applications. If higher energy gamma rays are needed, sources utilizing nuclear reactions are available. As it turns out, the (α, n) neutron sources are also good sources of high energy gamma rays. In the process of producing a neutron, an excited nucleus is also produced which in turn decays by gamma-ray emission. The gamma rays emitted in these processes usually appear in the energy range of 3-7 MeV. However, the accompanying neutrons may pose both a biological problem as well as an unwanted type of radiation in the intended use of the source.

Annihilation radiation is produced by the interaction of a particle with its anti-particle. The most common example of this is electron-positron annihilation. In this process, the electron and positron combine to produce two photons each with an energy equal to the rest mass of the electron ($E_e = m_e c^2 = 0.511$ MeV). These annihilation photons are emitted in opposite directions. The inverse of this process exists when a gamma ray interacts with an electromagnetic field to create an electron positron pair. This is called pair production (see Section 1.3).

Another form of electromagnetic radiation of interest is bremsstrahlung (braking radiation) and is the process used in the production of x rays from conventional x-rays machines. This process takes place when an electric charge is decelerated as it slows down or stops when interacting with matter. The fraction of kinetic energy converted into bremsstrahlung increases with increasing electron energy and is largest for absorber materials of higher atomic number (high Z). Bremsstrahlung is also produced by beta particles, therefore some bremsstrahlung is generated by any encapsulated isotope which undergoes beta decay.

Characteristic x rays are produced from atomic electron transitions. These transitions usually occur in the K, L, and M electron shells of an atom (transitions among higher electron shells give rise to optical radiation). Characteristic x rays can originate from two different classes of processes, nuclear decay and external.

The term characteristic refers to the fact that the x-ray spectra produced by each element is unique to that element. This fact can be exploited to determine unknown elements in a sample bombarded with electrons.

The K-series x rays have the highest energy. Their energy increases for elements with higher atomic number (Z). For example, for elements with a Z near 10, K-series x rays are produced with energies about 1 keV, increase to about 10 keV around $Z = 30$, and to about 100 keV around $Z = 90$. The characteristic x-ray spectra of the elements is well known and can be found in the Table of Isotopes (Lederer, 1967).

Although a K shell vacancy can be filled by a free or unbound electron, the vacancy is normally filled by an electron from a higher orbital. This is called cascading and results in a series of characteristics x rays from the higher orbital transitions.

X-ray emission from an excited atom is a competing process with auger electron emission. In this case an electron from a higher shell is ejected and the vacancy in the lower shell is filled by another electron from the higher level. This results in two vacancies in the higher level which are filled by still higher level electrons. The energy of the auger electron from the L shell filling a K shell vacancy is the difference in energy between the K x-ray energy and the binding energy of the L shell. Auger electron emission is favored in low- Z elements (for which electron binding energies are small) and therefore they have relatively low energies. The fraction of all cases in which an excited atom emits a characteristic x ray instead of an auger electron is defined as the fluorescence yield and is usually tabulated with other spectroscopic data.

There are predominantly two processes originating in the nucleus that result in the production of characteristic x rays, electron capture and internal conversion. Electron capture is a process competing with positron emission (e^+). Instead of emitting a positron, the nucleus captures an orbital electron (usually a K shell electron). This capture process creates a vacancy in the atomic shell which is filled with the generation of characteristic x rays and/or auger electrons. Electron capture may produce a daughter nucleus in an excited state which will in turn decay by gamma-ray emission. There are however, some radionuclides that undergo electron capture to the ground state of the daughter nucleus. In this case, only the characteristic x rays are emitted. An example of this is iron-55, it is very nearly a pure source of manganese K-series x rays at about 5.9 keV. In the process of internal conversion, defined earlier in this section, the ejection of an orbital electron leaves a vacancy that is again filled with the production of characteristic x rays and/or auger electrons. In internal conversion the energy of the excited nucleus which would otherwise appear as a gamma ray is instead transferred directly to one of the orbital electrons (typically K or L shell). This electron is ejected from the atom with an energy found by subtracting the binding energy

of the orbital electron from the nuclear excitation energy. These electrons are called conversion electrons and are nearly monoenergetic.

Characteristic x rays are created by external processes in a variety of ways involving every form of ionizing radiation. There are advantages and disadvantages associated with each method and type of ionizing radiation used. However, the energy of the x ray emitted depends only on the target material. The higher the atomic number (Z) the higher the x-ray energy emitted.

Characteristic x rays created by high energy electron bombardment will be accompanied by the bremsstrahlung continuum produced by the interaction of the electrons with the target material. Although the bremsstrahlung radiation will be emitted preferentially in the forward direction, the characteristic x rays will be emitted isotropically.

Characteristic x rays can also be created using alpha emitters. The advantage to this technique is that there is no bremsstrahlung radiation produced. Providing a pure alpha emitter is chosen, a clean characteristic x-ray spectra can be produced.

1.2.3 Electrons and Beta Radiation

The term beta (β) radiation is a general term for fast electrons (e^-) or positrons (e^+) emitted during a radioactive decay process. Beta particles are usually produced with endpoint energies ranging from a few keV to MeVs. There are however, other sources of energetic electrons, namely auger electrons, conversion electrons and electron accelerators. The distinction between electrons and betas is that fast electrons are produced by processes acting on the electron shell of an atom and betas (β) are produced in the radioactive decay of one of the nucleons in a nucleus.

The process of beta decay is an isobaric process which occurs in an unstable nucleus with an excess neutron. During the decay, a neutron in the nucleus is converted to a proton, a beta particle and an anti-neutrino with the latter pair sharing most of the excess energy. Each specific beta decay transition is characterized by a fixed energy or Q-value. Anti-neutrinos have such extremely small interaction probabilities with matter that they are undetectable for all practical purposes. The beta particle formed in the decay of a particular nucleus can range from zero to the "beta endpoint energy" which is equal to the Q-value. A typical beta energy spectrum is shown in Figure 1.1.

Most beta decays produce a daughter nucleus in an excited state which in turn emits one or more gamma rays. There are however, "pure beta emitters" which decay directly to the ground state of the daughter as shown in Figure 1.1. The excited daughter can also decay by a process of Internal Conversion. The beta particle may decay to one of several excited states in the daughter nucleus and each excited level can result in conversion electrons with different

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energies coming from different atomic orbital levels. Consequently, the combined beta and conversion electron spectra can be quite complicated.

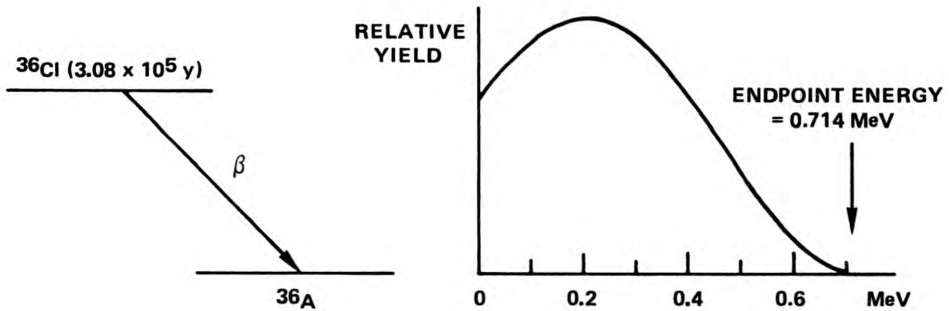


FIGURE 1.1 THE DECAY SCHEME OF ^{36}Cl AND THE RESULTING BETA PARTICLE ENERGY DISTRIBUTION.

The vacancies left in electron shells by the conversion electrons or by electron capture are often filled by electrons from higher orbital levels. The resulting transitions generate x rays which are characteristic of the specific states involved. A simple conversion electron spectrum is shown in Figure 1.2. The hole left in the electron shell can be filled by a process resulting in auger electrons rather than x rays.

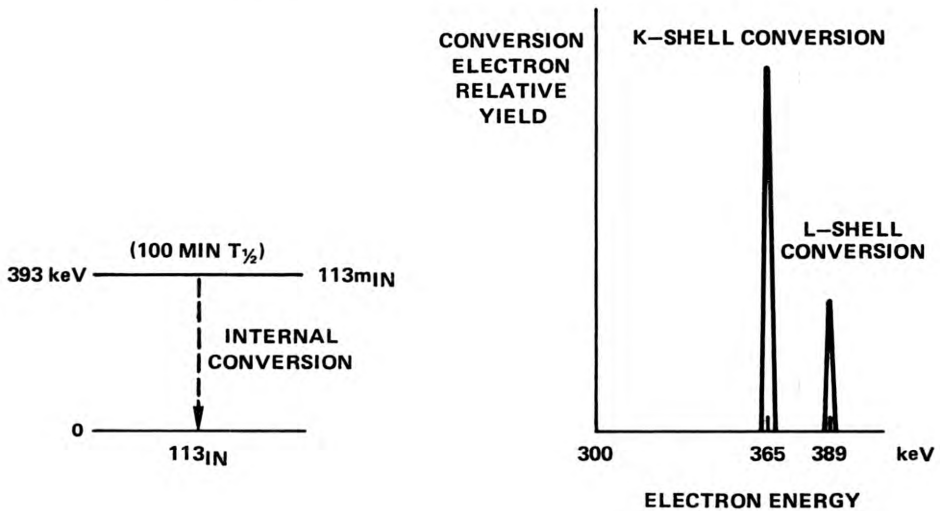


FIGURE 1.2 THE CONVERSION ELECTRON SPECTRUM EXPECTED FROM INTERNAL CONVERSION OF THE ISOMERIC LEVEL AT 393 keV IN $^{113\text{m}}\text{In}$.

The most intense source of fast electrons is the electron accelerator. There are many types of electron accelerators. Generally they work in the following manner. A metal filament is heated to boil off electrons. These electrons are thermal electrons

and have energies less than 1 eV. The thermal electrons are then accelerated by several high voltage electrodes in series, each having a higher electric potential than the previous one. The energetic electrons so produced can be focused by magnetic fields and electrostatic lenses to produce a highly collimated, fairly monoenergetic beam of high energy electrons. Most laboratory scale accelerators are capable of producing electrons with energies up to about 100 keV. For higher energies, a large scale accelerator is required. These machines are called linear accelerators or linacs. They are capable of producing electrons with energies up to several GeV.

1.2.4 Heavy Charged Particles

When a nucleus decays by heavy charged particle emission, the decay energy (Q-value) is shared between the heavy particle and the parent (recoil) nucleus. This is a consequence of conservation of energy and momentum. The heavy particles will have less energy than the Q-value of the transition. For example, alpha particles appear with an energy given by $Q(A-4)/A$, where Q is the Q-value of the transition and A is the mass number of the parent nucleus.

Alpha particles are doubly ionized helium ions. In the radioactive decay process they can be produced in one or more energy groups and for all practical purposes each group can be considered monoenergetic. Most radioisotope sources of alpha particles emit these in the energy range between 4 and 6 MeV. This results from a strong relation between the half-life of the parent nucleus and the energy of the alpha produced in the decay. Radionuclides that emit energies higher than 6 MeV have such short half-lives that they are not practical for use as sources. On the other hand, radionuclides that emit alphas with energies less than 4 MeV have such long half-lives that their specific activity (i.e., activity per unit mass) is so low that they are not practical either.

The only radioisotope source of heavy charged particles with masses greater than the alpha particle is the spontaneous fission source. Naturally occurring spontaneous fission only occurs in the very large mass number, i.e., uranium-235, plutonium-237, etc., actinide series and transuranic isotopes like californium-252 and curium-244. Each fission gives rise to two fission fragments, an average of 2 to 3 neutrons, and fission gamma rays. Isotopes which can decay by spontaneous fission usually decay by alpha emission and only rarely by fission. Many of the fission fragments that are produced are also unstable and may undergo decay. Consequently, the spontaneous fission source emits every type of radiation and can be packaged to produce one or all types of radiation by a selection of the proper casing and backing material. A further discussion can be found in Section 1.3.

Fission fragments are medium-mass positive-ions with typical mass and energy distributions as shown in Figure 1.3. There are

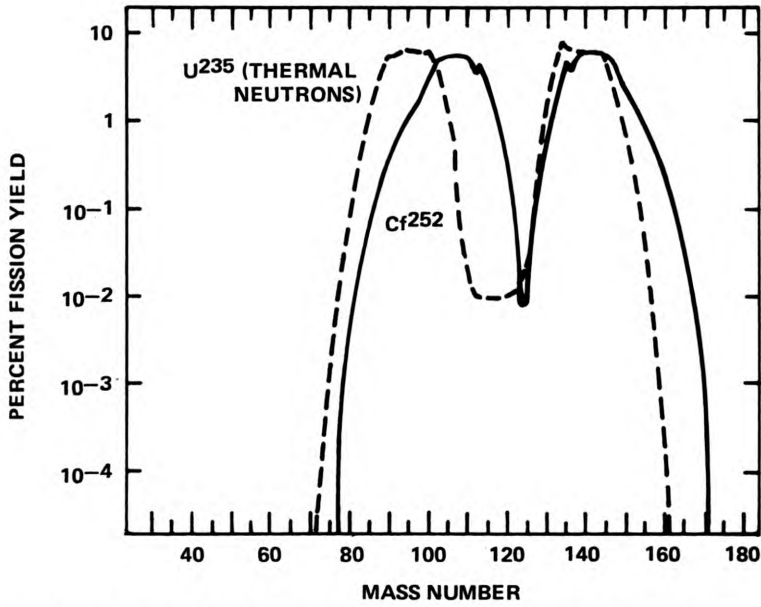


FIGURE 1.3 (a) THE MASS DISTRIBUTION OF ^{252}Cf SPONTANEOUS FISSION FRAGMENTS. ALSO SHOWN IS THE CORRESPONDING DISTRIBUTION FROM FISSION OF ^{235}U INDUCED BY THERMAL NEUTRONS (NERVIK, 1963).

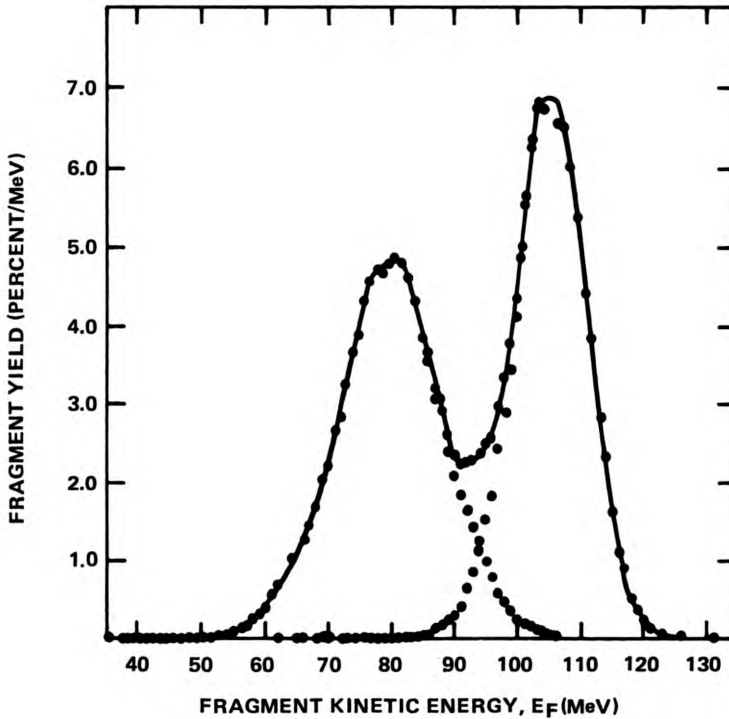


FIGURE 1.3 (b) THE DISTRIBUTION IN KINETIC ENERGY OF THE ^{252}Cf SPONTANEOUS FISSION FRAGMENTS. THE PEAK ON THE LEFT CORRESPONDS TO THE HEAVY FRAGMENTS, AND THAT ON THE RIGHT TO THE LIGHT FRAGMENTS (WHETSTONE, 1963).

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usually two groups of fission fragments formed in the fission process, the "light group" and the "heavy group" with average mass numbers of 108 and 143 respectively. The fragments appear initially as positive ions with a net charge as high as 18 or 22. As the fragments interact with matter, they slow down and additional electrons are picked up. The fragments come to rest as neutral atoms having released about 185 MeV of energy in the process.

Energetic heavy charged particles can be artificially produced in machine sources called particle accelerators. Almost all designs are based on producing ions with low energy and accelerating them to high energy by either a dc electric field or an alternating electric or magnetic field. These sources are typically very expensive and are confined to research laboratories.

1.2.5 Neutrons

Free neutrons (those not bound in a nucleus) are unstable. They undergo beta decay to form a proton with a 12 minute half-life. Most neutron producing reactions and decay schemes also produce gamma rays. Consequently, neutron radiation is usually accompanied by gamma radiation (there is often sufficient shielding that betas are not present for practical purposes).

The most intense source of neutrons is the nuclear explosive. Neutron fluxes on the order of 10^{23} n/cm²/sec can be produced. However, nuclear reactors are the most intense conventional sources of neutrons. Neutron fluxes can be produced with almost any intensity up to a practical limit of about those produced in weapons. The neutrons produced in a reactor come from a variety of sources including photoneutrons and decay neutrons, but primarily are produced by fission.

Neutrons are responsible for producing the chain reaction in a reactor. On the average 2 to 3 neutrons are produced in the fission process. Of these neutrons at least one must interact with another fuel nucleus (e.g., uranium-235) to continue the chain. The other neutrons are usually absorbed in a non-fissioning process or escape from the reactor. The reactor power level is controlled by increasing or decreasing the neutron growth rate in the core.

One type of source with an appreciable half-life is the spontaneous fission source mentioned in the previous section. A common spontaneous fission source is californium-252. Its half-life of 2.65 years is long enough to be used as a laboratory source and is one of the most widely produced transuranic elements. Although the dominant decay mechanism is alpha decay, a significant number of neutrons are also produced (during spontaneous fission). In fact, 2.3×10^6 n/sec are produced per microgram of the sample. The neutrons so produced appear in the MeV range (1-10 MeV).

Widely used laboratory neutron sources are the (α ,n) and (γ ,n), i.e. photoneutron, sources. These sources are made by mixing an alpha or gamma emitter with a radionuclide that has a large

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cross section for the (α, n) or photoneutron reaction.

The ideal (α, n) neutron source is made from an alloy of an actinide element (alpha emitter) and beryllium. A stable alloy of beryllium and an actinide metal in the form of MBe_{13} (where M represents the actinide metal) can be formed and has the appropriate concentration ratio of actinide and beryllium. It is desirable to have the alpha emitter present in a small relative concentration. In this configuration, only one alpha in 10^4 reacts with a beryllium nucleus to produce a neutron. The ^{239}Pu -Be source is one of the most widely used (α, n) neutron sources. However, since about 16 grams of the alloy is required for 1 curie (3.7×10^{10} Bq) of activity, sources of this type of a few centimeters in dimension are limited to a neutron production rate of about 10^7 n/sec. Sources that utilize shorter lived actinides (e.g., americium-241 and plutonium-238) can produce higher neutron yields but have shorter lifetimes. Other alpha emitters such as actinium-227 and radium-226 produce even higher neutron yields but are accompanied by high gamma-ray emission rates. These sources are not practical for most applications where an intense gamma-ray background is undesirable.

The neutron energy spectra (in the MeV energy range) from all such alpha-Be sources are similar, and any differences reflect the small differences in alpha energy. For larger sources, the secondary processes of neutron scattering, $(n, 2n)$ reactions in beryllium, and $(n, \text{fission})$ reactions with the actinide can introduce some dependence of the energy spectrum on the source size (Anderson, 1972).

To insure that the actinide-beryllium alloy remains intact (for safety precautions), the material is usually doubly encapsulated in individually welded stainless steel capsules. The inner capsule containing the alloy is somewhat smaller than the outer capsule to provide an expansion volume for the helium gas produced when the alpha particles are stopped and neutralized.

The decay rate of the (α, n) source usually decays with the decay rate of the alpha emitter except in cases where the daughter of the alpha emitters is also an alpha emitter or where there may be other radioisotopes present. For example, many ^{239}Pu -Be sources contain other isotopes of plutonium as well. The isotope plutonium-241 beta-decays to americium-241, an alpha emitter. In this case, the neutron source emission rate will actually increase with time as the americium-241 concentration builds up.

Some gamma ray emitting radioisotopes can also be used to produce neutrons when combined with an appropriate target material and are called photoneutron (γ, n) sources. Only two target nuclei have low enough photoneutron thresholds to have any practical value in photoneutron sources, namely, beryllium-9 and deuterium (^2H). There are some advantages and disadvantages encountered in photoneutron sources compared with other types of neutron sources. A definite advantage of the photoneutron source is that if the gamma source is monoenergetic the neutrons produced for a given energy gamma will be essentially monoenergetic, although with the larger sources

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neutron scattering will produce an energy spread. The main disadvantage of photoneutron sources is the large gamma-ray background that accompanies neutron sources of appreciable intensity. It turns out that only one gamma ray in 10^5 or 10^6 interacts with a target nuclei to produce a neutron. Another disadvantage is that many of the required gamma emitters (e.g., sodium-24) have such short half-lives that they must be re-activated in a nuclear reactor between uses.

Radioisotope photoneutron sources are usually constructed in such a way that the gamma emitter is encapsulated by a hollow aluminum sphere and coated on the outside with the target material, beryllium or deuterium.

Photoneutron production can be carried out on a much larger scale with the use of electron accelerators. In this case high energy electrons are bombarded on a target to produce MeV bremsstrahlung photons that can be used to interact with an appropriate isotope to produce neutrons. A variety of photoneutron reactions are possible including (γ, n) , $(\gamma, 2n)$, and $(\gamma, \text{fission})$ reactions producing a wide variety of neutron energies. Other neutron sources that can be made from accelerated charged particles can range in size from a compact neutron generator consisting of a sealed tube containing the ion source and target, together with a portable high voltage generator, to large machine sources such as cyclotrons and Van de Graaff accelerators. Two of the most common reactions used in these types of sources are the D-D and D-T fusion reactions (D and T are isotopes of hydrogen). Neutrons produced in the D-D reaction appear with energies near 3 MeV while D-T neutrons appear near 14 MeV. A one milliamp beam of deuterons will produce about 10^9 n/sec from a deuterium target and about 10^{11} n/sec from a tritium target.

1.3 INTERACTION OF RADIATION WITH MATTER

1.3.1 General Considerations

In this section, a discussion of how radiation interacts with matter is presented. First, the general terms and concepts will be discussed, then the characteristics of each type of radiation will be treated in separate subsections.

When radiation interacts with matter, there are a multitude of microscopic processes which play a role in producing macroscopic changes in the material. These effects include ionization (radiation detection), energy production (fission), transmutation of elements (radioisotope production), and radiation damage (cancer therapy). Although radiation damage in biological matter can be used for desirable effects, it is the undesirable effects on man and his environment that are of ultimate concern.

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For the purpose of discussing how different forms of radiation interact with matter it is convenient to present the various types of radiation in two categories: charged particle radiation and uncharged radiation.

Charged particles like electrons, alphas, and fission fragments interact continuously through the coulombic force with the electrons present in any medium through which they pass. In addition, the heavy charged particles (protons, alphas, and fission fragments) have a very short range in condensed matter (on the order of 10^{-5} meters) because of their large mass. Electrons on the other hand have a much larger range in matter (on the order of 10^{-3} meters).

Uncharged radiations like x rays and neutrons are not subject to the coulomb force. Instead, this radiation interacts with matter by first undergoing a catastrophic interaction which radically alters the properties of the incident radiation in a single encounter. For photons, the interaction usually results in the full or partial transfer of energy of the incident particle to electrons or nuclei of the constituent atoms. Consequently they have a much greater penetration power in matter (on the order of 1/10 meter). Another feature of uncharged radiation is that through the interaction process with matter, other types of (secondary) radiation may be produced. For example, an x or gamma ray can interact with an electron transferring most or all of its energy. The resulting electron has all of the characteristics of a beta particle and its corresponding interaction properties. This electron loses energy by interacting with matter to produce ion pairs. In contrast to photons, neutrons interact with nuclei to produce (in most cases) heavy charged particles.

Devices designed to detect various types of radiation take advantage of their interaction properties and the secondary particles produced to generate an output signal.

When discussing the individual microscopic processes that occur when radiation interacts with matter, it is convenient to treat each form of radiation as a beam of particles (although the wave nature of radiation also plays a role) interacting with individual atoms or electrons (or groups of them).

There are two basic interaction mechanisms that occur when a beam of radiation passes through matter, i.e., absorption and scattering. In the process of absorption, the incident particle disappears producing another type of particle (or particles). Examples of absorption events are fission (neutron absorption) and pair production (gamma or x-ray absorption). There are two types of scattering events, elastic and inelastic scattering. Elastic scattering occurs when an incident particle strikes another particle and transfers some or all of its kinetic energy to the other particle. Inelastic scattering occurs when an incident particle strikes another particle and some of its kinetic energy is converted into excitation energy which in turn may be radiated away. In this case the incident particle loses a greater fraction of energy than it

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would in an equivalent elastic collision.

The probability that an incident particle will interact with a target nuclei or electron can be expressed as a cross section (σ) in units of area (cm^2). A special unit of cross section is the barn ($1 \text{ barn} = 10^{-24} \text{ cm}^2$). To better understand the concept of the cross section it is instructive to examine the following experiment. Suppose a beam with intensity I is incident on a given volume of matter. In the volume there are a number of nuclei, N . Many of the incident particles will pass through the volume without interacting with any of the nuclei. The number of particles which do collide is found to be proportional to the incident beam intensity I and the number of nuclei N in the path of the beam. These observations can be summarized by the following equation:

$$\text{Number of Collisions per Second} = \sigma IN \quad (1.1)$$

where σ , the proportionality constant called the cross section, I is the incident intensity, and N is the total number of particles in the path of the incident beam. The number of interactions per second with a single nucleus is therefore just σI . It follows that σ is equal to the number of interactions per second with one nucleus per unit intensity of the beam.

The above equation can be rewritten as:

$$\text{Number of Collisions per Second} = In\sigma V \quad (1.2)$$

where I is the intensity ($\text{particles}/\text{cm}^2/\text{sec}$), n is the number density of the target nuclei ($\text{nuclei}/\text{cm}^3$), σ is the cross section (cm^2), and V is the volume of the target (cm^3). It follows from the above equation that the number of collisions/ cm^3/sec in the target, which is called the collision density F , is given by $F = In\sigma$.

There may be many ways in which a particle may interact with target nuclei. In the case of neutron interaction there are scattering interactions with a cross section (σ_s) and absorption interactions such as radiative capture (σ_γ) and fission (σ_f). The total interaction cross section is expressed as the sum of the component cross sections, i.e., $\sigma_t = \sigma_s + \sigma_\gamma + \sigma_f$.

The product of the atom density n and a cross section (σ) occurs frequently in the equations of radiation physics. It is given a special symbol, Σ , and is called the macroscopic cross section. Since n and σ have the units of cm^{-3} and cm^2 respectively, Σ has the units cm^{-1} and is the probability per unit path length that an interaction will occur.

The reciprocal of Σ is called the mean free path. This quantity, which is designated by the symbol l , is equal to the distance that an incident beam of radiation will be reduced to e^{-1} (i.e., 0.368) of its original value due to interaction between the beam and the material through which it is passing.

1.3.2 X ray and Gamma ray Interactions

There are three principle interaction mechanisms for x and gamma rays with matter: photoelectric absorption, Compton scattering, and pair production. Photoelectric absorption and Compton scattering are interactions between gamma rays and electrons, whereas pair production are interactions between photons and the electric field of a nucleus (or infrequently with the electric field of an atomic electron).

The fundamentals of x and gamma-ray interaction mechanisms are introduced here, but are again reviewed in Section 1.5 in the context of their influence on the response of gamma-ray detectors.

Photoelectric Absorption

In the photoelectric absorption process an incoming gamma ray interacts with an atom in such a way that the photon is completely absorbed and an energetic photoelectron is ejected from the atom.

The photoelectron appears with an energy given by $E_e = h\nu - E_b$ where E_b represents the binding energy of the photoelectron in its orbital shell. As mentioned in Section 1.2.2, once an orbital electron has been ejected from an atom a characteristic x ray or auger electron result.

The photoelectric process is the predominant mode of interaction for x and gamma rays of relatively low energy (below 1 MeV). The process is also enhanced for absorber materials of high atomic number Z . Consequently many x and gamma-ray shielding materials are made of high Z elements such as lead.

Compton Scattering

The interaction process of Compton scattering takes place between an incident x or gamma-ray photon and an electron in the absorbing material. It is the predominant interaction mechanism in tissue-like material for photon energies typical of radioisotope sources.

In Compton scattering, the incoming photon interacts with an atomic electron considered to be free and is deflected through an angle Ω with respect to its original direction. The photon transfers some of its energy to the electron. The energy transferred to the electron can vary from zero to a large fraction of the incident photon energy and depends heavily on the scattering angle Ω . The expression which relates the energy transfer and the scattering angle along with a graphic representation of the interaction is shown in Figure 1.4.

For small scattering angles Ω , very little energy is transferred. Even in the extreme case of $\Omega = 180^\circ$ (back-scatter), some of the original energy is retained by the incident photon. Figure 1.5 shows the variation of scattered photon energy with scattering angle. As shown, any scattering angle greater than about $110-120^\circ$ results in scattered photons of nearly identical energy

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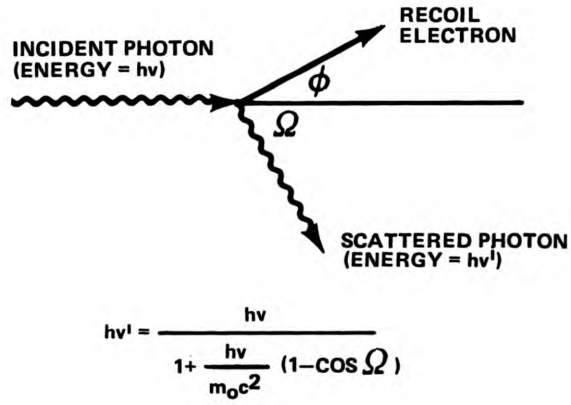


FIGURE 1.4 COMPTON SCATTERING EVENT.

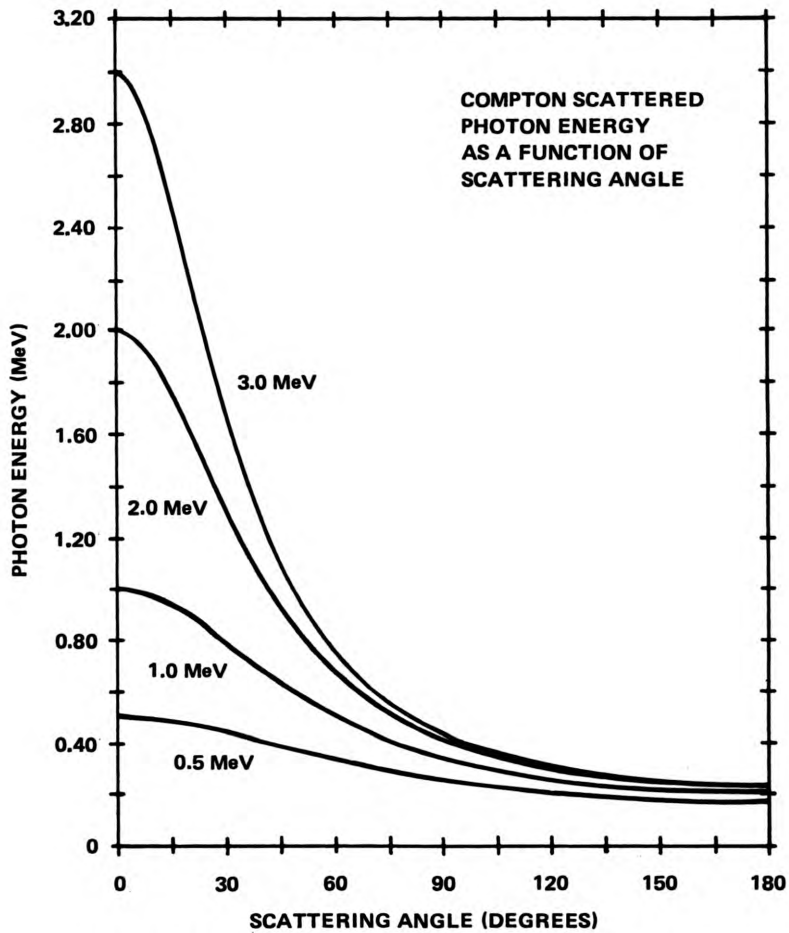


FIGURE 1.5 VARIATION OF SCATTERED GAMMA RAY ENERGY WITH SCATTERING ANGLE (KNOLL, 1979).

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(about 0.2 MeV) regardless of the incident photon energy. This is commonly referred to as the "backscatter" peak in gamma-ray spectrometry.

The probability of Compton scattering increases with the number of electrons available as scattering targets and therefore increases linearly with elements of higher Z . The dependence of Compton scattering on photon energy generally decreases with increased photon energy (see Figure 1.6). On the other hand, the higher the incident photon energy, the stronger the tendency for forward scattering and thus less of the higher energy photon's energy is transferred to a recoil electron.

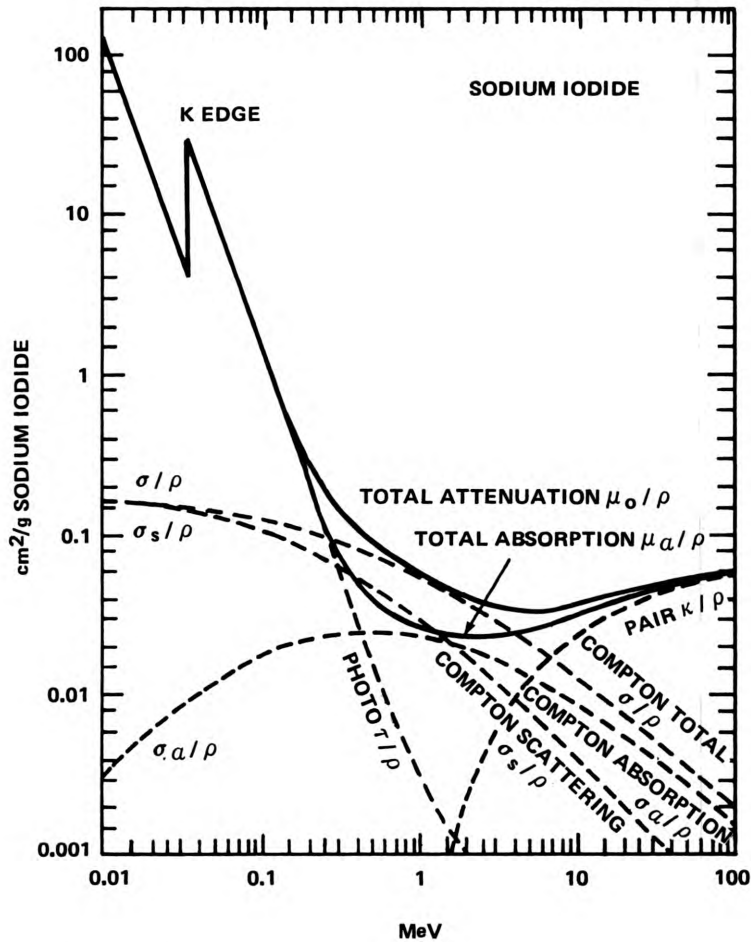


FIGURE 1.6 ENERGY DEPENDENCE OF THE VARIOUS GAMMA RAY INTERACTION PROCESSES IN SODIUM IODIDE. (AFTER EVANS, 1955).

Pair Production

If the x or gamma ray photon energy exceeds twice that of the rest mass of an electron (e.g., 2×0.511 MeV), the process of pair production is energetically possible. In fact, for high energy photons, pair production is the predominant interaction mechanism with matter.

Pair production interactions must take place in a coulomb field. The probability of such an event increases approximately with the square of the absorber atomic number (Z^2). In the interaction, the incident photon is absorbed and an electron-positron pair is produced. All of the energy in excess of the 1.02 MeV required to produce the pair appears as kinetic energy which is shared by the positron and electron. As mentioned in Section 1.2.5, once a positron is created it interacts with an electron to produce annihilation radiation. Therefore, annihilation radiation always accompanies pair production.

Attenuation Coefficients

The microscopic interaction events of absorption and scattering produce the macroscopic events of attenuation and extinction in a beam of radiation. Attenuation of incident radiation occurs when absorption and scattering events reduce the amount of radiation originally present. This process is exponential in nature and can be described by the following equation:

$$I(x) = I_0 e^{-\mu x} \quad (1.3)$$

where I_0 is the incident intensity, $I(x)$ is the intensity at a given distance x through material with a linear attenuation coefficient μ . A graphic description of simple exponential absorption is shown in Figure 1.7.

It should be pointed out that in this description of exponential attenuation the incident radiation is assumed to be a highly collimated beam of monoenergetic radiation and that any interaction which changes the direction (i.e., scattering) of an incident radiation is considered to remove it from the beam.

As mentioned earlier, there are three interaction mechanisms which can remove a gamma-ray photon from the beam either by absorption or scattering. These processes can be characterized by a probability of occurrence per unit path length in the absorber. The sum of these probabilities is the probability per unit path length that a gamma ray is removed from the beam and is called the linear attenuation coefficient μ , where $\mu = \tau(\text{photo}) + \sigma(\text{compton}) + k(\text{pair})$. Figure 1.6 shows the energy dependence of the various gamma-ray interaction processes in Sodium Iodide (a common detector medium).

The mean free path, l , for gamma rays in an absorbing medium is the reciprocal of the linear attenuation coefficient μ . Typical

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values of λ range from a few mm to tens of cms in solids for common gamma-ray energies.

Use of the linear attenuation coefficient is limited by the fact that it varies with the density of the absorber. Therefore, the mass attenuation coefficient, defined as μ/ρ where ρ is the density of the medium, is more widely used. For a given gamma-ray energy, the mass attenuation coefficient does not change with the physical state of a given absorber. For example, it is the same for water whether present in solid, liquid, or vapor form. The mass attenuation coefficient of a compound or mixture of elements can be calculated as the sum of the weighted fractions of each element multiplied by its own mass attenuation coefficient.

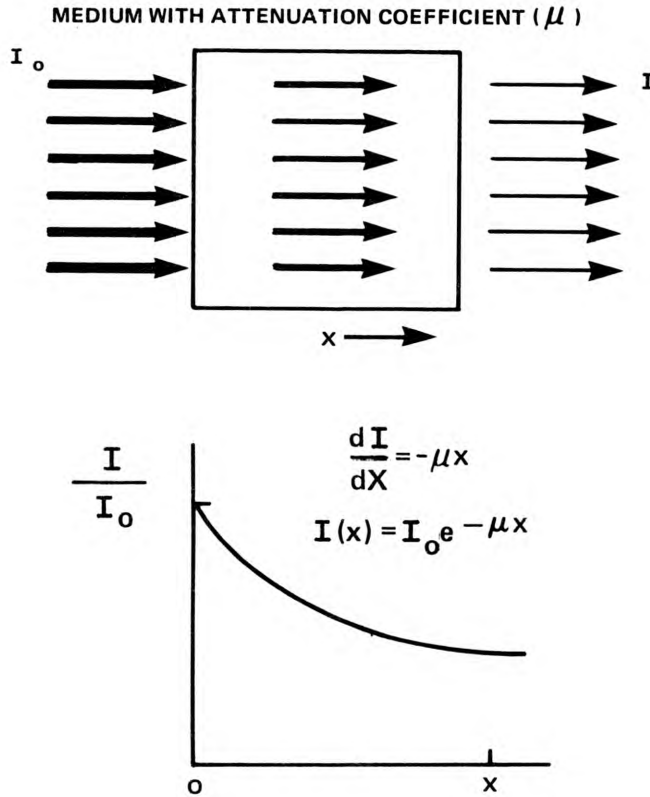


FIGURE 1.7 SIMPLE EXPONENTIAL ATTENUATION

The attenuation law for gamma rays can now be written in the form:

$$I(x) = I_0 e^{-\left(\frac{\mu}{\rho}\right)\rho x} \quad (1.4)$$

The product ρx is known as the mass thickness of the absorber. The unit of mass thickness is g/cm^2 or in SI units kg/m^2 . The thickness of absorbers used in radiation measurements is often

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measured in mass thickness rather than linear thickness.

Buildup Factor

The process of gamma-ray attenuation was described earlier treating the incident radiation as a highly collimated beam. The essential characteristic here is that only gamma rays from the source which escape interaction entirely in the absorber can be counted by the detector. This is sometimes characterized as a "narrow beam" or "good geometry" measurement (See Figure 1.8a). However, measurements are usually carried out under conditions where the incident radiation beam is not collimated (Figure 1.8b).

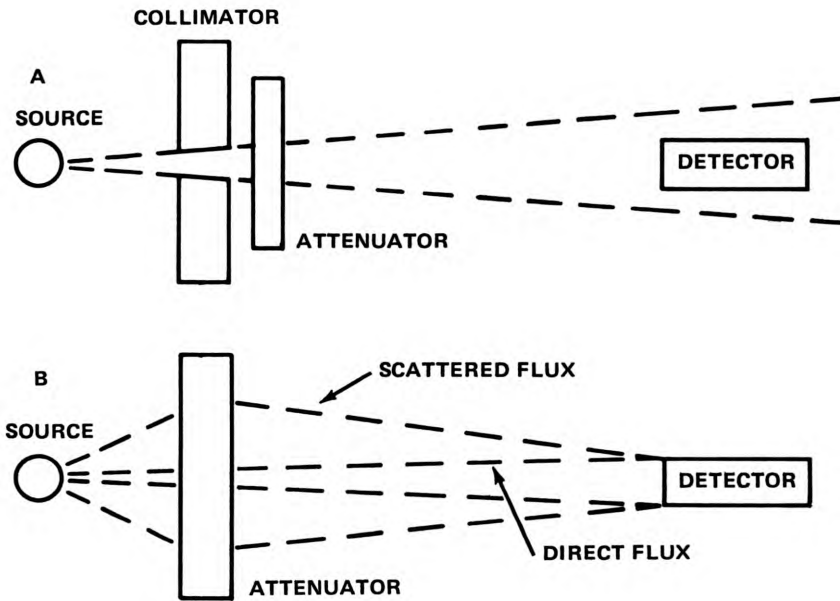


FIGURE 1.8 SHIELDING GEOMETRY. (A) IS GOOD NARROW BEAM GEOMETRY. (B) IS POOR OR BROAD BEAM GEOMETRY.

The detector shown in Figure 1.8b can either respond to gamma rays directly from the source, or to gamma rays that reach the detector after having been scattered. The conditions that lead to simple exponential attenuation are violated under these conditions because of the added contribution of the scattered gamma rays. The equation for simple exponential attenuation is modified for this situation by adding a buildup factor such that:

$$\frac{I}{I_0} = B(x, E_\gamma) e^{-\mu x} \quad (1.5)$$

The buildup factor $B(x, E_\gamma)$ is a function of absorber thickness and gamma-ray energy. The magnitude of the buildup factor depends on the type of detector used. It also depends on the specific geometry

of the setup. As a rough rule of thumb, the buildup factor for thick slab absorbers tends to be about equal to the thickness of the absorber measured in units of the mean free path, l , of the gamma rays, provided that the detector can respond to a broad range of gamma-ray energies.

1.3.3 Beta Radiation Interactions

Beta particles, like all other charged particles, interact in a continuous manner with the coulombic field of atoms and molecules present in any absorbing medium. Energy is transferred from the beta particle to the atoms in the medium resulting in ionization and excitation. However, beta particles, unlike the heavy charged particles may also lose an appreciable amount of energy by radiative processes (bremsstrahlung). When compared with heavy charged particles, betas lose energy at a lower rate and follow a more tortuous path through materials before they expend their kinetic energy and come to rest. These paths are due to the multiple scattering events that occur with atoms of the absorbing medium.

The way in which beta particles lose energy in an absorbing medium can be characterized by a linear energy loss (linear stopping power) defined as the quotient dE by dx where dE is the increment of energy loss traversing a distance dx . Since electrons lose energy, the linear energy loss has a negative sign. There are two components of the energy loss for betas, namely, collisional losses and radiative losses. Thus, the total linear stopping power for beta particles in an absorbing medium is the sum of the collisional and radiative specific energy losses. A more common characterization of the energy loss is the total mass stopping power. This is the total linear stopping power divided by the density of the media.

The energy loss due to collision is to a first approximation proportional to Z/A of the material in which the electrons are slowing down. There is a slight energy dependence through a β^{-2} term as well as a logarithmic term involving β^2 (where $\beta = v/c$).

The energy loss of the betas due to radiative losses is proportional to the electron kinetic energy and the square of the absorber atomic number (actually this latter dependence is $Z(Z + 1)/A$). Thus the radiative losses are most important for high electron energies and for absorber materials of large atomic number. Since most sources of beta particles produce them with energies less than a few MeV, radiative losses are a small fraction of the energy loss due to collisions, and are significant only in absorber materials of high atomic number.

If the reciprocal of the total stopping power is integrated from zero to the initial beta energy, one obtains an expression for the range. Berger and Seltzer (1982) have calculated the ranges and stopping powers of electrons and positrons using the continuous slowing down approximation. This "csda" range then is the total length of the path of a hypothetical electron that dissipated all of

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its energy continuously at the mean rate, as given by the stopping power. For a given energy the csda range (in units of g/cm^2) increases roughly as Z/A . A power law can be fitted as a function of energy for a given element which will be valid over a limited energy region.

The "practical" range for monoenergetic electrons can be defined as the absorber thickness required to reduce beta particle penetration to the background level. The absorption curve for monoenergetic beta particles has a linear portion. If this linear portion is extrapolated to zero, as is done in Figure 1.9, this extrapolated range is found to be characteristic of a given beta energy and is usually taken as the practical range. Most beta particle sources produce a continuous spectrum of energies, and as can be seen in Figure 1.10, the range in this case is quite indeterminate. The reason for this is that the soft or low energy beta particles are rapidly absorbed even in small thicknesses of the absorber, so that the initial slope of the attenuation curve is much greater. For the majority of beta spectra, the curve is approximately exponential in shape and is therefore nearly linear when plotted on semi-log coordinates as in Figure 1.10.

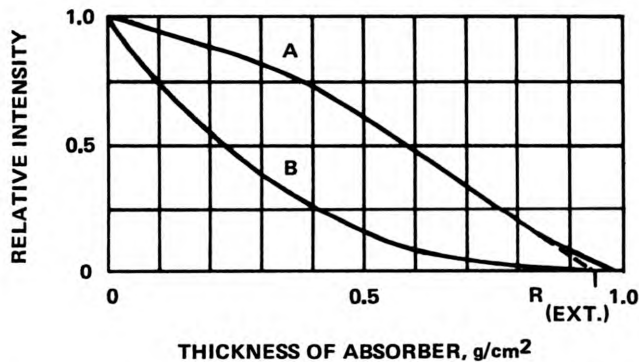


FIGURE 1.9 ABSORPTION CURVES FOR (CURVE A) MONOENERGETIC ELECTRONS (1.9 MeV) AND (CURVE B) BETA PARTICLES WITH CONTINUOUS ENERGY SPECTRUM ($E_{\max} = 1.9$ MeV) (PRICE, 1963).

Another feature associated with beta particle interactions with absorbing materials is backscatter. A beta particle entering one surface of an absorber may undergo a large-angle deflection and re-emerge from the surface through which it entered. These backscattered beta particles do not deposit all of their energy in the absorbing medium, and therefore the energy fluence inside the media will be less than expected. If the source has a thick encapsulation material, the backscattered betas will not contribute to the source strength. Backscattering is enhanced for low energy betas and absorbers with high atomic number.

Most beta-active sources can be shielded with a few centimeters of shielding material. However, if the shielding material consists of high Z material like lead, the exposure created by the source wi

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actually increase due to bremsstrahlung production by the betas and the fact that bremsstrahlung has a much greater range than similar energy beta particles. Therefore, it is not a good practice to shield beta active sources with lead or other high Z materials unless there is sufficient thickness to attenuate the bremsstrahlung as well.

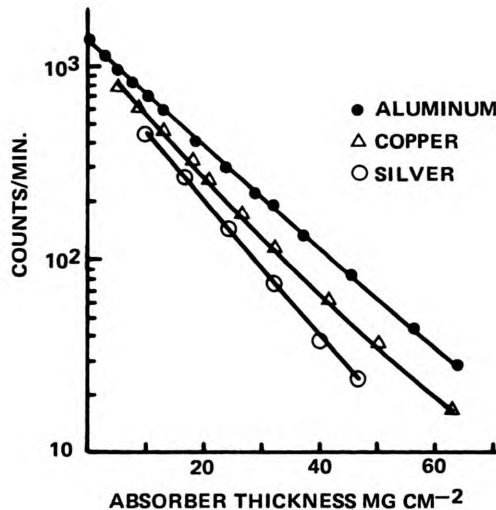


FIGURE 1.10 TRANSMISSION CURVES FOR BETA PARTICLES FROM ^{185}W (ENDPOINT ENERGY OF 0.43 MeV) (BALAKMENS, 1977).

Positrons

Positrons interact in much the same way as do other beta particles. The magnitude of the coulomb force is the same whether the interaction involves an attractive or repulsive force since the mass of the positron and electron are the same. However, at the end of the range of a positron, annihilation radiation (mentioned in Section 1.2.5) is generated. These 0.511 MeV annihilation photons have a much greater penetrating power compared to the range of the positron. Therefore, the energy deposition characteristics are somewhat different.

1.3.4 Heavy Charged Particle Interactions

Heavy charged particles interact with matter primarily through the coulombic field present in an absorbing medium. Most interactions occur with the orbital electrons within of the absorbing medium, although interactions with nuclei are also possible. Rutherford scattering and (α, n) reactions are examples of interactions with nuclei.

The heavy charged particle may remove an atomic electron from

its shell (ionization) or may only raise the electron to a higher orbital level (excitation). In some instances (in particularly close encounters) an electron may be given sufficient impulse so that after having left its parent atom, it may still have enough kinetic energy to create further ionizations. These energetic electrons are called delta rays. The energy required to ionize and excite atoms in the absorbing medium reduces the kinetic energy of the heavy charged particle causing it to slow down.

It can be shown (from the conservation of momentum and energy) that the fraction of kinetic energy that a particle can transfer to another in a collision is related to the ratio of the masses of each of the participating particles. A particle colliding with another particle of equal mass can transfer most (or all) of its kinetic energy. The concept is analogous to impedance matching in an electrical circuit. (A power source can deliver the maximum energy to a load whose impedance is equal to that of the power source.) Consequently, heavy charged particles can transfer very little (about $1/500$ for alphas) of their energy to a single electron. However, due to the large momentum each heavy particle has, it follows a very straight path except at the end. Heavy charged particles are therefore characterized by a definite range in a given absorber material. The "practical" range, as defined in the previous section, represents a distance (measured parallel to the incident trajectory) beyond which no particles will penetrate.

The interactions between heavy charged particles and absorbing materials can be characterized by a linear stopping power S , and a range. The classical expression that describes the energy loss is known as the Bethe formula. Some generalizations can be made from this expression to show how the energy loss varies as a function of a few key parameters, namely, the density and Z of the absorber, and the charge state and kinetic energy of the charged particle. In general, the energy loss increases with absorber density and atomic number Z of the media and with the square of the charge of the incident particle. It decreases with increased particle energy. This can be heuristically explained by noting that the higher the electron density (ρZ), the more electrons a particle must interact in traversing a given distance. Also, the higher the charge of the incident particle, the greater the electric field strength which increases the interaction forces. Finally, the faster an incident particle travels, the shorter the time interval it spends in the vicinity of a given electron, thus reducing the impulse felt by that electron. Therefore, when comparing heavy charged particles with the same velocity, the particles with the greatest charge will lose energy at the fastest rate. However, heavy charged particles like fission fragments with initial charges around $+20e$ have a complex energy loss process because as they slow down they immediately begin to reduce their charge by picking up electrons. Consequently, in contrast to other heavy charge particles like alphas and protons, fission fragments actually have a lower stopping power as they slow down because the decrease in the effective charge offsets the

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increase in stopping power that normally accompanies slower moving particles.

In order to describe the concept of particle range (for heavy charged particles) it is instructive to refer to a conceptual experiment shown in Figure 1.11. Here a collimated source of monoenergetic alpha particles is counted by a detector after passing through an absorber of variable thickness. The ratio of the detected number of alphas (I) to the number detected without the absorber (I_0) is plotted as a function of absorber thickness t . For small values of absorber thickness, the only effect is to cause an energy loss of the alphas in the absorber as they pass through. The energy loss ΔE can be calculated from $\Delta E = -St$, where S is the stopping power averaged over the energy of the particle in the absorber of thickness t . Since the tracks through the absorber are straight, the total number of alphas that reach the detector remains the same. No attenuation in the number of alphas takes place until the thickness of the absorber approaches the range of the alphas (as shown in Figure 1.11). Not all of the particles in a beam of monoenergetic charged particles behave in exactly the same manner so there are small fluctuations in the path length of a beam of particles which is known as range straggling. Hence, when the thickness of the absorber reaches the length of the shortest track, attenuation of the beam begins and as the thickness increases, the intensity of the detected beam drops rapidly to zero.

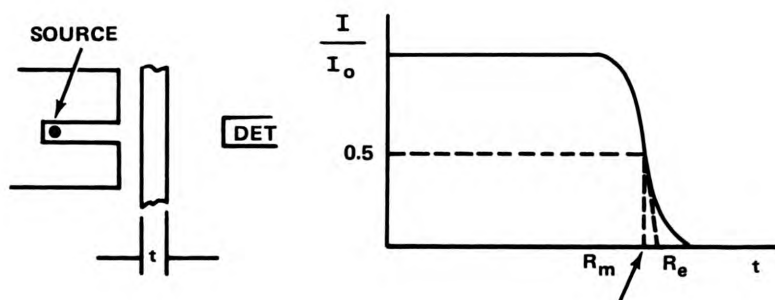


FIGURE 1.11 AN ALPHA PARTICLE TRANSMISSION EXPERIMENT. I IS THE DETECTED NUMBER OF ALPHAS THROUGH AN ABSORBER THICKNESS t , WHEREAS I_0 IS THE NUMBER DETECTED WITHOUT THE ABSORBER. THE MEAN RANGE R_m AND EXTRAPOLATED RANGE R_e ARE INDICATED (KNOLL, 1979).

The relationship between alpha particle range and incident energy in various absorbing media is shown in Figure 1.12. Units of the range are given as usual in mass per unit area to minimize the differences in the individual curves.

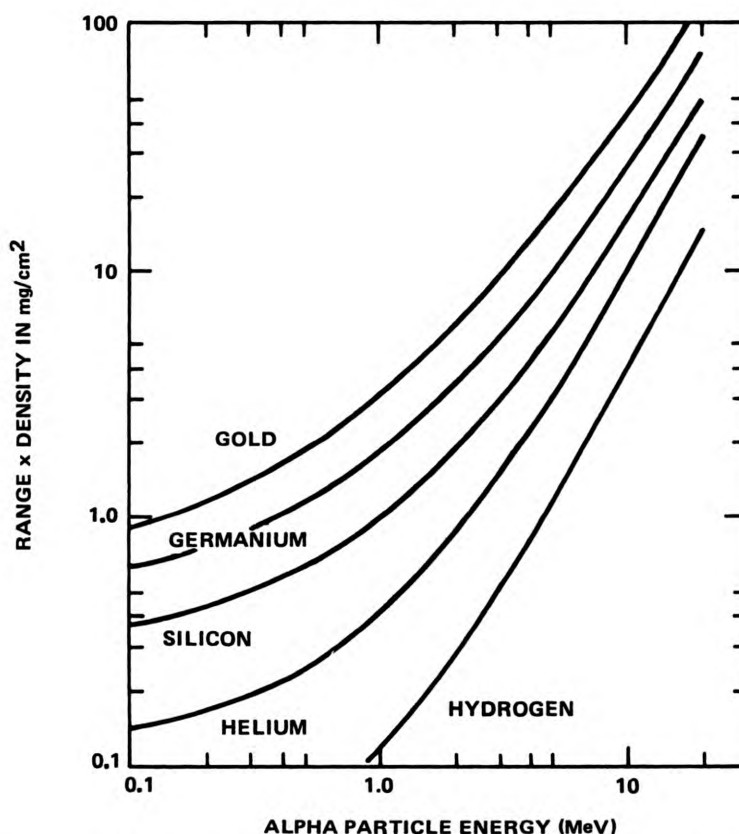


FIGURE 1.12 RANGE-ENERGY CURVES CALCULATED FOR ALPHA PARTICLES IN DIFFERENT MATERIALS. UNITS OF THE RANGE ARE GIVEN IN MASS THICKNESS TO MINIMIZE THE DIFFERENCES IN THESE CURVES (WILLIAMSON et al., 1966).

1.3.5 Neutron Interactions

Neutrons, like all uncharged radiations, do not interact with matter through the coulombic force which dominates the energy loss mechanisms for charged particles and electrons. Therefore, neutrons can travel through many centimeters of matter before an interaction takes place. Neutrons interact exclusively with the nuclei in the absorbing medium.

There are many different types of reactions between neutrons and nuclei. They can be classified by either scattering events or by nuclear reactions.

The probability (cross section) that a given reaction will occur varies as a function of neutron energy. To aid in the discussion of interactions at various neutron energies, neutrons are classified into two categories on the basis of their energy.

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Neutrons with energies below 0.5 eV are called "slow" neutrons and those with energies above 0.5 eV are called "fast". As a further classification, neutrons which have slowed down to the point where they are in thermal equilibrium with ambient temperature are called "thermal neutrons." At room temperature neutrons have an average energy of 0.025 eV and have an average velocity of 2200 meters/second.

Fast Neutron Interactions

Most neutron sources produce neutrons whose energies are in the MeV range. The most important interaction mechanisms for neutrons in this energy range are elastic and inelastic scattering events.

Fast neutrons can transfer an appreciable amount of energy to a recoil nucleus in one collision. The amount of energy is dependent upon the mass of the recoil nucleus and to some extent the scattering angle. Neutrons can transfer up to all of their kinetic energy to a hydrogen nucleus (see Section 1.3.4) in a single collision. Hence, some fast neutron detectors utilize this phenomenon and are called proton recoil detectors (see Chapter 3).

If the energy of the fast neutron is sufficiently high, inelastic scattering with heavier nuclei can take place in which the recoil nucleus is elevated to one of its excited states during the collision. The nucleus subsequently de-excites by emitting a gamma ray. In this case the neutron loses a greater fraction of its energy than it would in the case of elastic scattering. The slowing down process in which neutrons transfer kinetic energy to absorber nuclei is called neutron moderation.

There are a number of fast neutron reactions with various nuclei. For example uranium-238 will undergo fission only with fast neutrons. However, the cross sections for these reactions are quite small (a few barns).

Slow Neutron Interactions

Fast neutrons that have been moderated by scattering events slow down and thermalize with the absorbing medium. These slow neutrons have significant interaction probabilities with absorbing nuclei. In fact, most absorption cross sections are inversely proportional to the neutron velocity. This phenomenon is referred to as a $1/v$ characteristic (where v is the neutron velocity).

This $1/v$ behavior of neutron absorbers creates another phenomenon called spectrum hardening. If a thermal (Maxwell-Boltzmann) neutron spectrum is incident on an absorbing medium, the lower energies are preferentially absorbed, shifting the mean energy of the spectrum to a higher energy.

There are a number of neutron induced nuclear reactions that are possible, e.g., (n,α) , (n,γ) , (n,p) , and $(n,fission)$. These reactions play an important role in neutron shielding and detection. The wide variety of possible neutron reactions are characterized by

their respective cross sections and are plotted as a function of energy in a publication by Garber and Kinsey (1975).

1.4 RADIATION QUANTITIES AND UNITS

1.4.1 Activity

Radioactivity occurs when an unstable nucleus undergoes transformation to a more stable state. In the transformation process energy is released and is carried away from the nucleus in the form of charged particles, neutrons, neutrinos, or electromagnetic radiation. There is naturally occurring radioactivity as well as that which is artificially induced.

The activity (A) of a radioisotope source is defined as its rate of decay, and for a radioisotope decaying to a stable daughter is given by

$$A = \frac{dN}{dt} = -\lambda N \quad (1.6)$$

where N is the number of radioactive nuclei and λ is the decay constant. The historical unit of activity is the curie (Ci) defined as 3.7×10^{10} disintegrations per second. The SI unit is the becquerel (Bq). The conversion between units is $1 \text{ Ci} = 3.7 \times 10^{10} \text{ Bq}$. This is the activity of one gram of pure radium-226.

The activity of a radioactive source is not always equivalent to the emission rate of radiation produced by its decay. In a given decay scheme several photons may be emitted (e.g., Co-60 beta decays to an excited level in Ni-60 which almost always de-excites by a cascade process emitting two gamma rays). Therefore, a knowledge of the decay scheme of a particular isotope must be known before activity can be inferred from emission rate.

The half-life of a radioisotope is the time required for half of the nuclei originally present to decay. In the next half-life, one half of the remaining nuclei will have decayed and so forth. Consequently, after five half-lives, only $1/32$ or about 3 percent of the original nuclei will remain. The half-life of a given radioisotope source is related to its decay constant (λ) by the following equation:

$$t_{1/2} = \frac{\ln 2}{\lambda} = \frac{0.693}{\lambda} \quad (1.7)$$

The various isotopes that are formed during radioactive decay can be determined by using the Chart of the Nuclides. A small section of the chart is shown in Figure 1.13. The Chart of the Nuclides is a listing of all of the known isotopes laid out on a grid according to

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their mass numbers in a coordinate system whose vertical axis is Z (atomic number) and horizontal axis is N (number of neutrons). Thus, the isotopes of a given chemical element (same Z) are lined up horizontally and isobars (same A) are lined up diagonally, left to right. In Figure 1.13 the daughter product of any given form of radioactive decay can be found by following the arrows indicated for each type of decay from the original nucleus to its corresponding daughter.

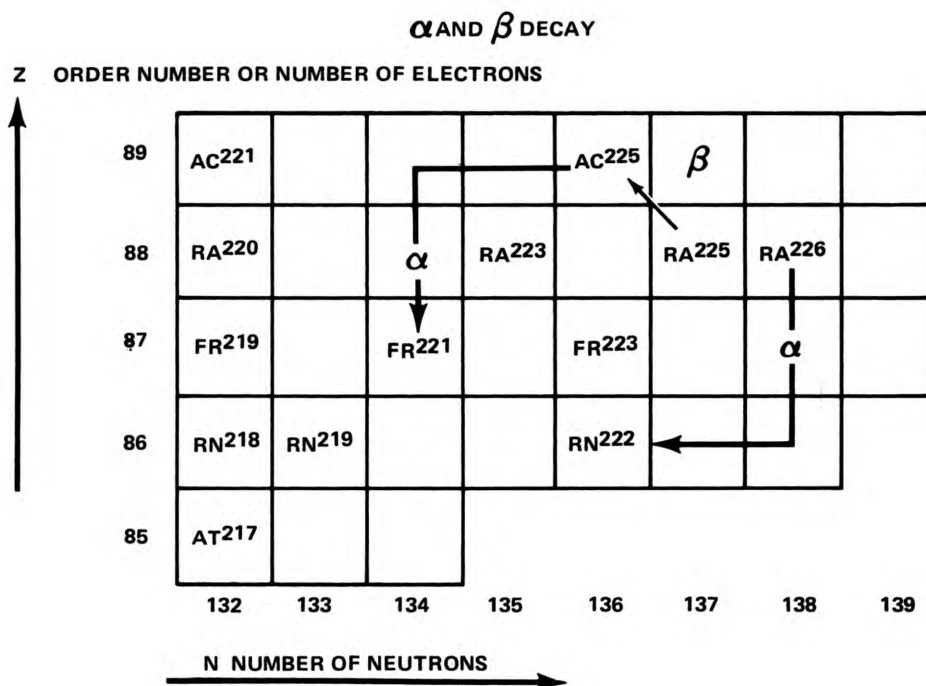


FIGURE 1.13 DISPLACEMENT OF NUCLEI BY RADIOACTIVE DECAY

Many isotopes like uranium-238 have successive radioactive daughters that in turn decay, until finally, a stable isotope is formed. In the case of uranium-238, there are 13 radioactive daughters in the decay chain and the resulting stable isotope is lead-204.

The rate of change of the daughter in the chain, dN_2/dt , is equal to the rate of production of the daughter minus the rate of decay of the daughter. Since the former is simply the rate of decay of the parent, i.e., the activity of the parent, A_1 , is

$$A_1 = N_1\lambda_1 = N_0\lambda_1 e^{-\lambda_1 t} = A_0 e^{-\lambda_1 t} \quad (1.8)$$

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and the rate of growth of the daughter is

$$\frac{dN_2}{dt} = N_1\lambda_1 - N_2\lambda_2 \quad (1.9)$$

For no daughters initially present, the solution to this equation is

$$N_2 = \frac{\lambda_1 N_0}{\lambda_2 - \lambda_1} (e^{-\lambda_1 t} - e^{-\lambda_2 t}) \quad (1.10)$$

and the corresponding activity of the daughter is

$$A_2 = N_2\lambda_2 \quad (1.11)$$

Note that the activity of the daughter is $N_2\lambda_2$ and not dN_2/dt . If the daughter has a shorter half-life than the parent ($\lambda_2 > \lambda_1$), the activity of the daughter can be expressed as

$$A_2 = \frac{\lambda_2}{\lambda_2 - \lambda_1} A_1 [1 - e^{-(\lambda_2 - \lambda_1)t}] \quad (1.12)$$

Since $(\lambda_2 - \lambda_1)$ is positive for this case, the ratio of the daughter to parent activity approaches a constant value of $\lambda_2/(\lambda_2 - \lambda_1)$ for long times. This constant ratio of activity between parent and daughter nuclei is referred to as transient equilibrium. For the special case where the daughter half-life is much shorter than the parent ($\lambda_2 \gg \lambda_1$)

$$A_2 \approx A_1 (1 - e^{-\lambda_2 t}) \quad (1.13)$$

If t is greater than several half-lives of the daughter, $A_2 \sim A_1$. This is referred to as secular equilibrium and is a special case of transient equilibrium. Since $A_2 = N_2\lambda_2$, the number of daughter nuclei in secular equilibrium is

$$N_2 = \frac{\lambda_1}{\lambda_2} N_1 = \frac{\lambda_1}{\lambda_2} N_0 e^{-\lambda_1 t} = \frac{A_0 e^{-\lambda_1 t}}{\lambda_2} \quad (1.14)$$

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The number of daughter nuclei present after secular equilibrium occurs depends on the decay constant of the parent. The activity of the daughter, $N_2\lambda_2$, appears to be decaying with the same half-life as the parent.

Figure 1.14 shows the case for a daughter with a decay constant 10 times that of the parent. The activity is plotted on a log scale. The activity of the daughter builds up to time t_e . At t_e the activity of the daughter is maximum. After time t_e both the daughter and parent appear to be decaying with the half-life of the parent since the activities are now in a constant ratio and transient equilibrium exists. In this case the ratio of the activities is 1.1.

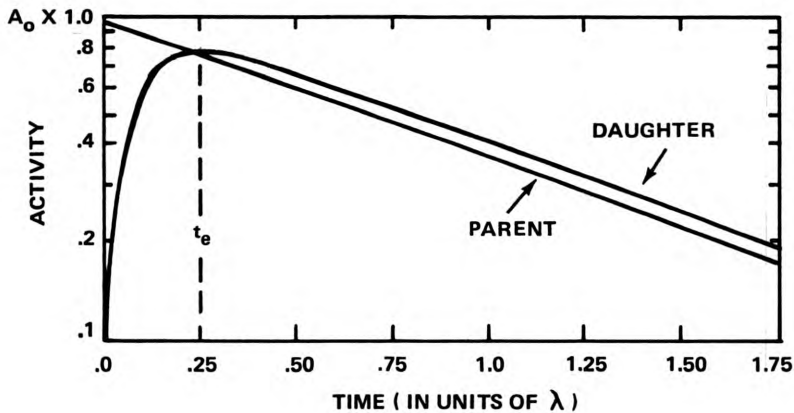


FIGURE 1.14 ACTIVITY OF PARENT AND DAUGHTER RADIONUCLIDE FOR CASE OF λ OF DAUGHTER EQUAL 10 λ OF PARENT.

1.4.2 Exposure, Absorbed Dose, Dose Equivalent

Exposure

The special unit of exposure in air is called the roentgen (abbreviated R) and is defined in ICRU 33 as "the quotient of ΔQ by ΔM , where ΔQ is the sum of the electrical charges on all the ions of one sign produced in air when all the electrons, liberated by photons in a volume element of air whose mass is ΔM , are completely stopped in air" ($1 \text{ R} = 2.58 \times 10^{-4} \text{ coulombs/kg}$). In this definition, the ionization arising from the absorption of bremsstrahlung emitted by electrons is not to be included in ΔQ .

Figure 1.15 shows several of the important considerations in understanding the definition of exposure. The mass of air of interest is that inside the circle. Here photon 1 interacts inside the mass. The electron produced in the interaction (dashed line) begins to lose its energy by producing ion pairs (dots). For photon 1, the electron is completely stopped inside M. All of the charge produced along the electron's path is to be counted in determining

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the total charge. Note, however, along the path the electron also loses energy by radiating some bremsstrahlung (short wavy line). If this bremsstrahlung interacts inside M, any charge resulting from this interaction is not included in determining the total charge.

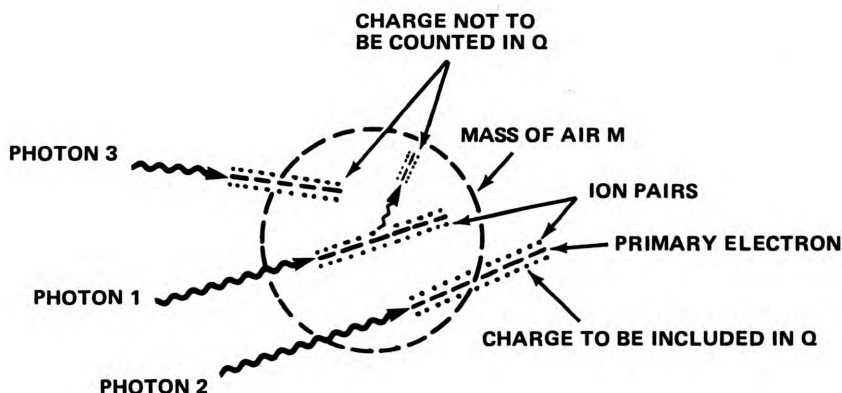


FIGURE 1.15 SCHEMATIC OF PHOTONS INTERACTING WITH A GIVEN MASS OF AIR SHOWING WHICH ELECTRON-ION PAIRS SHOULD BE COUNTED IN DETERMINING THE EXPOSURE.

Next consider photon 2. It also interacts inside M. This time not all of the charge produced by the slowing down electron is inside M. Thus in determining the total charge we have to add an amount to correct for the charge outside of M. Finally, consider photon 3. This interacts outside of M, but the range of the electron is such that part of the charge is produced inside of M. In evaluating the total charge produced by photons interacting inside of M, this charge must be subtracted from the sum inside of M. If the amount of charge not collected inside of M by photon 2 is just compensated by the extra charge deposited inside of M by photon 3, charge particle equilibrium is said to exist. Since it takes 33.85 electron-volts to produce an ion pair in air, an electron with an initial energy of 1 MeV will produce about 30,000 ion pairs in stopping.

Absorbed Dose

If instead of wanting to know the total amount of charge produced within a given mass of air, one were interested in the total amount of energy absorbed by the mass of air, one would have the absorbed dose to air. Instead of air one might be interested in how much energy some other material absorbs per unit mass. Thus when specifying absorbed dose, the material of interest should be specified.

ICRU 33 defines absorbed dose as "the quotient of dE by dm , where dE is the mean energy imparted by ionizing radiation to matter of mass dm . $D = dE/dm$." The special unit of absorbed dose is the rad which is $0.01 \text{ J/kg} = 100 \text{ erg/g}$. The SI unit of absorbed dose is 1 J/kg

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= 100 rads. The SI unit has been given a special name, the gray (Gy), 1 Gy = 1 J/kg.

Absorbed dose, for air, D_a , and exposure, X , are simply related by the mean energy expended per ion pair,

$$\bar{x} = \frac{W}{e} D_a \quad (1.15)$$

The absorbed dose in another medium, m , is related to the absorbed dose to air by the ratio of the mass-energy attenuation coefficients,

$$D_m = \frac{(\mu_{en}/\rho)_m}{(\mu_{en}/\rho)_a} D_a \quad (1.16)$$

Contrary to exposure, absorbed dose is defined for all types of ionizing radiation (exposure is only defined for photons, i.e., x or gamma rays).

The value for W/e of 33.85 eV per ion pair converts to 0.873 rad/R. For tissue and air, the ratio of the mass-energy attenuation coefficients is about 1.10 from about 100 keV to a few MeV. Thus to within about 5 percent the absorbed dose to tissue in rads is approximately numerically equal to the exposure in roentgens.

Although exposure rates are generally measured by free air or air-wall ionization chambers, the exposure corresponding to an absorbed dose can be measured by "tissue-equivalent" ionization chambers. These chambers are built with a wall material whose radiation absorption properties are similar to that of tissue and operate by satisfying the criteria of the Bragg-Gray principle. According to the Bragg-Gray principle, the amount of ionization produced in a small gas-filled cavity surrounded by a solid absorbing medium is proportional to the energy absorbed in the solid. There are certain restrictions implicit in this principle, i.e., the gas cavity must be small enough relative to the mass of the absorber to leave the energy fluence distributions of the primary electrons produced in the medium unchanged. This requirement is fulfilled if the primary electrons lose only a small fraction of their energy in traversing the gas-filled cavity. If the cavity is surrounded by a solid medium of sufficient thickness to establish electronic equilibrium, then the energy absorbed per unit mass of wall material is related to the energy absorbed per unit mass of gas in the cavity by a ratio of the stopping powers of each medium.

Dose Equivalent

The rem is the unit of radiation dose equivalent and is an acronym for roentgen equivalent man. The dose equivalent, expressed in rem, is a product of two factors in the case of external radiation and a third factor in the case of internal radiation. The two factors

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that define the rem for external radiation sources is the absorbed dose to tissue (in rads) and a quality factor (QF). The rem in this case is a product of these two factors. In the case of internal radiation, a third factor is used to take into account how different radioisotopes concentrate in the body. This factor is called the distribution factor (DF).

The absorbed dose rate produced by a given radiation source depends on many factors including the type and energy of the decay products as well as a geometry factor. Some forms of radiation (e.g., neutrons and alphas) have been found to do more biologic damage than others (e.g., beta, x and gamma rays). When comparing the relative biological damage, it is found that the higher the rate of linear energy transfer (the amount of energy deposited per unit path length) of the radiation, the more effective it is in damaging the organism. Thus, for a given absorbed dose rate, different types of radiation produce different effects. This phenomenon has been described as the relative biological effectiveness (RBE) and is often defined as the ratio of the amount of 200 keV x rays required to produce a given effect to the energy of any radiation required to produce the same effect. The RBE must be taken into effect when calculating the rem dose. A quality factor (QF) has been defined to normalize the RBE of different radiation types. Table 1.1 shows the QF for various types of radiation.

TABLE 1.1 QUALITY FACTORS FOR VARIOUS TYPES OF RADIATION.

EXTERNAL EXPOSURE		INTERNAL EXPOSURE	
RADIATION	QF	RADIATION	QF
X, GAMMA RAYS	1	X, GAMMA RAYS	1
β (EXCEPT VERY LOW ENERGY)	1	β ($E > 0.03\text{MeV}$)	1
THERMAL NEUTRONS	3	β ($E < 0.03\text{MeV}$)	1.7
EPITHERMAL NEUTRONS ($E = .005\text{MeV}$)	(2-2.5)	α PARTICLES	10
FAST NEUTRONS ($E > .01\text{MeV}$)	(5-10)	OTHER HEAVY PARTICLES	20
HEAVY PARTICLES	20		
PROTONS	(1-3)		

1.4.3 Measurement Units

This section deals with the terms which are used throughout this handbook and which are defined in ICRU 33. It is important that one distinguishes between a quantity and a unit. A physical quantity

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characterizes a physical phenomenon in terms that are suitable for numerical specifications. For example, energy, time, length, and exposure are all physical quantities. A unit is a selected reference sample of a quantity. Every physical quantity may be expressed as the product of a numerical value and a unit.

Two systems of units are commonly used with radiation measurement, special and SI. For example, the curie, roentgen, rad, and rem are special units for the quantities activity, exposure, absorbed dose, and dose equivalent respectively. However, ICRU 33 recommends that the use of these special units be phased out by the year 1985 and that they be replaced by the appropriate SI units.

SI units are divided into three classes: base units, derived units, and supplementary units. The base units are meter, kilogram, second, ampere, kelvin, mole, and candela for the quantities, length, mass, time, electric current, thermodynamic temperature, amount of substance, and luminous intensity, respectively.

Derived units are made by combining base units according to the algebraic relations linking the corresponding quantities.

The current supplementary units are radian and steradian for the quantities plane angle and solid angle respectively.

Table 1.2 shows the most commonly used quantities along with their SI units and special units. This table is only a fraction of all of the recommended quantities and units recommended by the ICRU. For a complete discussion of these units refer to ICRU 33.

1.5 GENERAL PROPERTIES OF RADIATION DETECTORS

1.5.1 Introduction

Most radiation detectors derive their output signal from the interaction of a single particle or photon with matter. The detecting matter may be in solid, liquid, or gaseous form. In many situations the interaction with matter results in generation of charges, which are then sensed by electronic circuitry (ionization chambers, proportional counters, GM counters, solid state detectors, etc.). Other interaction may result in the generation of photons (scintillation detectors). The time required for the interaction to take place is very short (sub-pico seconds), while the time required to collect the charges can be fairly long (as long as a few milliseconds in the case of ionization chambers or as short as a few nanoseconds in the case of semiconductor detectors).

The collection of the charges generated can be accomplished in two different ways. The first one is the "current mode." Here the charges generated by many interactions are accumulated and time averaged. The result is an electrical current, the value of which is related to the number of interactions which took place during the time interval over which averaging took place. The second method is

TABLE 1.2 RADIATION QUANTITIES AND UNITS

QUANTITY	SI UNIT	NAME FOR SI UNIT	SPECIAL UNIT	RELATIONSHIP OF SI UNIT TO SPECIAL UNIT
ACTIVITY	RECIPROCAL SECOND (s ⁻¹)	BECQUEREL (Bq)	CURIE (Ci)	1 Bq = $\frac{10^{-10}}{3.7}$ Ci
ABSORBED DOSE	JOULE PER KILOGRAM (J·kg ⁻¹)	GRAY (Gy)	RAD (rad)	1 Gy = 100 rad
ABSORBED DOSE RATE	JOULE PER KILOGRAM-SECOND (J·kg ⁻¹ s ⁻¹)	GRAY PER SECOND (Gy·s ⁻¹)	RAD PER SECOND (rad·s ⁻¹)	1 Gy·s ⁻¹ = 100 rad·s ⁻¹
EXPOSURE	COULOMB PER KILOGRAM (C·kg ⁻¹)	—	RONTGEN (R)	—
EXPOSURE RATE	COULOMB PER KILOGRAM-SECOND (C·kg ⁻¹ s ⁻¹)	—	RONTGEN PER SECOND (R·s ⁻¹)	—
DOSE EQUIVALENT	JOULE PER KILOGRAM (J·kg ⁻¹)	SIEVERT * (Sv) *	REM (rem)	1 Sv * = 100 rem

* Recommended by ICRU and ICRP but not accepted as part of the SI at the time of writing.

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to collect and measure the charges generated by each interaction. Since the interactions are individual events a number of charge pulses are collected. This is called the "pulse mode." The number of pulses may be related to the intensity of the ionizing radiation, while the height of the pulse amplitude is proportional to the magnitude of the charges generated in the interaction and hence to the energy of the initiating event.

Since the collection of the charge requires some time, detectors operating in the pulse mode can only distinguish between two events as long as their interaction with the detector material is separated by a time greater than that required for collection of the charges generated. The minimum time between two consecutive pulses required for proper detection is called the "dead time." If two events are closer together than the dead time, there are, depending on the type of detector, two possibilities for response of the detector. The second event is either ignored by the detector, or its response is added to the response of the first particle interaction and both are therefore treated as though they were coming from one event. If the detector ignores the second event, it acts as if it were paralyzed. Therefore one refers to the two possibilities as paralyzable or non-paralyzable responses.

Another important property of radiation detectors is their efficiency. Efficiency is a measure of how many events will be recorded by a detector for a given set of experimental conditions. Several types of efficiency are of interest. The intrinsic efficiency is a measure of the probability that a particular radiation will interact with the detector to produce an event, i.e., the number of events detected to the number of photons or particles incident. For monoenergetic gamma rays incident on a detector with linear attenuation coefficient μ and thickness t , the intrinsic efficiency is $1 - e^{-\mu t}$. In practice, all detectors subtend a finite solid angle. Often one is interested in how many events will be detected from a point source for a detector a given distance away. This is the absolute efficiency, and is the product of the solid angle and the intrinsic efficiency. Finally, one often wants to know how the intrinsic efficiency varies with the energy of the radiation. This is referred to as the relative efficiency.

1.5.2 Detector Types

Gas Filled Detectors. These detectors are based on the interaction of the incident radiation with the gas in the detector. This interaction generates primarily ionization and excitation along the track of the particle. Gas filled detectors are the ionization chamber, the proportional counter, and the Geiger-Mueller counter. In these detectors the result of the particle-gas interaction is an electron-ion pair.

The ionization chamber is a detector which collects the electron-ion pairs generated by ionization produced in the detector

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gas. The proportional counter is a device which multiplies the number of electron-ion pairs by a certain factor before collecting them. The Geiger-Mueller counter, finally, is a device in which each electron-ion pair generated develops into an avalanche (Townsend avalanche) which has to be artificially terminated.

To put these characteristics in perspective, Figure 1.16 illustrates the different mechanisms of electron-ion pair collection. If an electron-ion pair is left to itself it will after a certain time recombine into a neutral atom or molecule. In order that the generation of an electron-ion pair be recorded as an event the electron-ion pair must be collected before recombination takes place. The collection is accomplished by attracting the electron-ion pair to electrodes by an electric field.

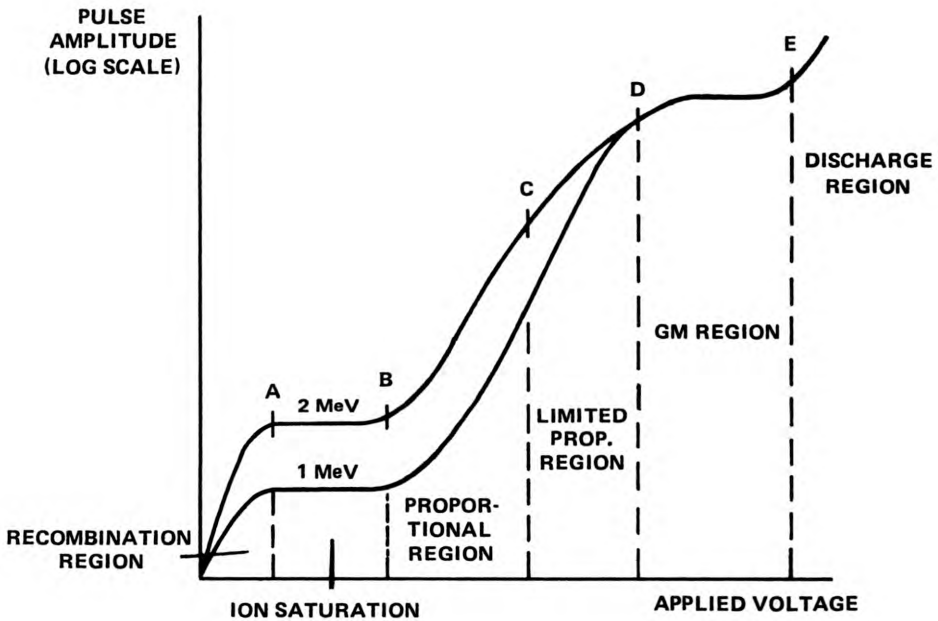


FIGURE 1.16 THE DIFFERENT REGIONS OF OPERATION OF GAS-FILLED DETECTORS. THE OBSERVED PULSE AMPLITUDE IS PLOTTED FOR EVENTS DEPOSITING TWO DIFFERENT AMOUNTS OF ENERGY WITHIN THE GAS.

For a parallel plate ionization chamber, the electric field strength (which determines the speed of collection) can be expressed as $E = V/d$ where V is the applied voltage and d is the electrode spacing. If the field strength is low, only a fraction of the electron-ion pairs will be collected while others will recombine. As the field strength is increased, more pairs will be collected and less will recombine. Finally, there is a field strength where all electron-ion pairs are collected. That corresponds to point "A" in Figure 1.16. Any increase in field strength will not collect any more electron-ion pairs. Therefore a plateau (point "A" to point "B") is reached. Since an ionization chamber is a device which

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collects electron-ions pairs, but does not multiply them, it is to be operated in the plateau region A-B.

If the field strength is increased beyond point "B" the electrons and ions are not only all collected, but they are also accelerated on their way to the collecting electrode with sufficient energy to produce more than one primary ion pair. The overall result is a multiplication of the original number of charge carriers. For a given field strength (in the region between B and D) the amount of current produced or the pulse amplitude is greater (but still proportional) than the energy lost by the initiating event. For the region B to C the pulse amplitude is proportional to the energy of the initiating event, while for the region C to D there is no longer a linear dependence. Past the point D the field strength applied is sufficient to cause an avalanche of secondary ions along the entire collecting anode. Each avalanche produced by the initiating event has enough energy to produce a cascade of avalanches resulting in a self-sustaining discharge. In principle one interaction event would start a discharge which would continue as long as the applied voltage is maintained. For counting purposes this is undesirable and one has to terminate the discharge artificially. This can be accomplished either by removing the applied voltage by appropriate circuitry or by adding a quenching agent to the gas. The quenching gas acts as an electron scavenger and terminates the discharge by removing electrons. GM tubes operate in this region. Since each event will produce a complete avalanche, there is no longer a relationship between the energy of the initiating event and the GM pulse amplitude. In fact GM tubes cannot distinguish between different types of radiation if the initial radiation can cause the avalanche to begin. Since there is an avalanche, there is again a voltage plateau region. At the end of the GM plateau, dielectric breakdown of the ionized gas begins. This is the discharge region E.

For gas filled detectors operated in a pulse mode, as long as the events are separated by a time greater than the pulse pair resolution time, the number of events will be proportional (with the efficiency) to the number of events incident on the detector. This is true for ionization chambers, proportional counters, and GM counters.

Scintillation Detectors. Certain materials emit visible or ultraviolet light when subjected to ionizing radiation. Radiation detection based on scintillation is one of the oldest methods used. The scintillation light is usually detected with a photomultiplier tube, although other light detecting media can be used. The material used as scintillators can be in solid, liquid, or gaseous form. Widely used materials are inorganic alkali halide crystals, plastics, silicate glasses, organic based liquids, and gases like Xe, Kr, and Ar.

Scintillation is a name which may cover both fluorescence and phosphorescence. In atomic physics it is usual to call the prompt emission of light of the same wavelength as the stimulating light

fluorescence, while phosphorescence would be the emission of longer wavelengths than the stimulating light. In case of solids the phosphorescence may also be delayed for appreciable time intervals. In the case of scintillations which are caused by ionizing radiation rather than light, the above definition no longer applies. Nevertheless, the two terms are used in this connection, mostly with the conotation that phosphorescence persists longer than fluorescence.

The mechanism of scintillation is similar to any mechanism which results in light emission. The requirement is that an atomic or molecular state has to be excited by the radiation to be detected, and has to decay to a lower state and consequently emit a photon.

The state may belong to a system of isolated states or it may be a part of a large number of degenerated states which form a conduction or valence band or an F-center (see Chapter 8).

Organic scintillators mostly possess isolated molecular states. These materials will therefore scintillate in any form, solid, liquid, or vapor. Inorganic materials like NaI rely on F-centers and need therefore to be in the form of a lattice structure to be able to scintillate. The noble gases, of course, have isolated atomic states.

Semiconductor Detectors. The density of atoms or molecules in a solid is several order of magnitude higher than in a gas. Consequently, for the same number of interactions, the active volume of a solid state radiation detector can be considerably smaller than that of a gas filled detector. Scintillation detectors are one example of solid state detectors. However, here light is generated instead of an electrical signal. Conversion of light to an electrical signal involves a certain efficiency which is by no means very large. This is in addition to the basic efficiency of the interaction of radiation with matter itself. Therefore, a solid state detector which does not have to go through the intermediate step of conversion of light to a electrical signal would be advantageous. Since early 1960 semiconductors which operate this way have become available for radiation detection.

The interaction of an incident particle with the active volume of a semiconductor detector results in creation of electron-hole pairs along the particle track. Such pairs may be compared to the electron-ion pair in a gas filled detector. They also are influenced by an applied electric field, but rather than being free charges which can be collected, injection of electron-hole pairs increase (or decrease) the number of charge carriers in one of the bands of the semiconductor and therefore increases the conductivity of the material. While charge carriers travel very slowly in a solid, the voltage pulse resulting from the stepped up conductivity travels fast. Therefore, in general, semiconductor detectors are very fast. They can be built very small and, in fact, are limited to relatively small sizes. Germanium and silicon are typical semiconductor materials used for radiation detectors.

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Other Detectors. While the gas-filled, scintillation, and semiconductor detectors are very widely used for radiation detection, there are other means to detect radiation. Such detectors may be used in the future or are already widely used (e.g. the thermoluminescent dosimeter). One very old detector which needs to be mentioned here is the photographic emulsion. It was and is still widely used for x-ray radiography in the form of x-ray film. Photographic emulsions are also used in film badge dosimeters. In the photographic emulsion the incident particle destroys a number of silver bromide or silver iodide molecules. This makes the affected photographic grain accessible to the reducing chemical agent, the developer, which if allowed to act long enough will reduce the whole grain to metallic silver. The unaffected particles will resist the developer and can be removed by a chemical agent, "the fix," which dissolves silver bromide, but not metallic silver.

A common type of personnel monitors are thermoluminescent dosimeters which are analogous to semiconductor detectors. However, thermoluminescent materials are generally insulators rather than semiconductors in that electron-hole pairs are generated by the incident radiation. Due to the poor conductivity, or in some cases due to low concentration of impurities (activators), recombination of the electron-hole pairs or injection of one type of charge carrier into a band does not take place at all, or takes place very slowly. When these dosimeters are heated, the traps can de-excite with the emission of photons which are then detected with a phototube. The number of photons emitted depends on the temperature to which the dosimeter was heated. A plot of light output versus temperature (the "glow curve") is used as a measure of the incident radiation (Chapter 8).

Another detector is the Cerenkov Detector. This detects light created when a charged particle enters the detector with a velocity greater than the speed of light in the detector medium. The effect can be compared to the sonic boom created by an airplane traveling faster than the speed of sound. Only charged particles are capable of producing Cerenkov radiation and only in a medium which has a refractive index substantially larger than 1. Gamma rays may be recorded by a Cerenkov detector if they create a Compton electron traveling with sufficient speed. Typical materials for Cerenkov detectors include water, liquid nitrogen, lucite, and glycerine.

In addition to the detectors mentioned above, there are still others, i.e., the spark detector, the track-etch detector, the cloud chamber, and the bubble chamber. There undoubtedly will be more discoveries in the future which may lead to new and better radiation detectors.

1.5.3 Detector Systems

While the detector itself is the heart of a radiation detection system, there are other components necessary to perform radiation detection measurements. These components are amplifiers, discriminators, converters (e.g. time to pulse height, analog to digital) and logic circuitry (e.g. to check for coincidence). One specific detector may be used with a variety of these components, and depending on the components chosen, the system may perform substantially different functions. The choice of components depends on whether the detector is operated in the current mode or pulse mode. Although in some cases the detector is operated in the pulse mode, the result is displayed on a rate meter making it similar to current mode operation.

If operated in the pulse mode, amplification and counting of pulses created by the incoming radiation is usually required. In addition, in most detectors - the GM counter being a notable exception - the height of the output pulse is related to the energy of the incident radiation. This allows spectroscopy of the observed radiation. This can be accomplished by sorting the observed pulses according to their height with the aid of a multichannel analyser. It is of course possible to use a single channel analyser and observe only pulses which correspond to a single band of energy. In other cases, where the pulses can be expected at the same time (e.g. electron-positron annihilation), one can check for prompt coincidence or delayed coincidence. An example of the latter is counting of fission fragments where the production of the fission fragments is signalled by a gamma photon and the fission fragment is detected some time later depending on its velocity and flight path. The delay time for the counting system can be varied and the amplifier for the detector can be gated for a certain time window. This way, the number of fission fragments having a given velocity can be counted.

For proper counting the observed pulse should be shaped so that it resembles a logic pulse (e.g., rectangular pulse). This can be accomplished with a discriminator. The logic pulse can be either counted and/or accumulated in a memory. It can also be integrated and displayed by a rate meter.

1.6 STATISTICS AND RADIATION MEASUREMENTS

1.6.1 Introduction

The measurement of nuclear radiation commonly involves phenomena which are statistical in nature. The understanding of these statistical effects is important, both from the standpoint of the design of calibration and measurement systems, as well as the interpretation of the results. The statistical nature of detection

systems is discussed first for pulse-type counters and then is extended to mean-level systems.

Statistical analysis is performed in nuclear radiation measurements to minimize the uncertainties due to the random nature of these processes. In addition, an apparatus can be tested for fluctuations due to causes other than statistical by comparison of the actual distribution of measurements results with that predicted by the statistical laws.

In addition to the normally occurring statistical fluctuations encountered in measurement systems, there are a series of errors that may be introduced into the system. Error is defined here as the difference between a calculated or observed value and the "true" value. Of course the "true" value in a given experiment is unknown, otherwise there would be no need to perform it. However, it is possible to know what the approximate value is either from prior experimentation or from theoretical calculations. Such approximations can yield an indication as to whether a result is of the right order of magnitude, but it is important to have some systematic way to determine from the data itself how much confidence one can have in its accuracy.

It is important here to distinguish between the terms precision and accuracy. The precision of an experiment is a measure of spread in the observed results without reference to what the central value of the result means. It is also a measure of how reproducible the result is. Absolute precision indicates the magnitude of the uncertainty in the result and is expressed in the same units as the result (e.g., 1.54 ± 0.02). Relative precision refers to the uncertainty in terms of a fraction of the value of the result (e.g., $1.54 \pm 1.3\%$). The accuracy of an experiment is a measure of how close the result of the experiment comes to the true value. Therefore, it is a measure of the correctness of the result, i.e., it is a combination of any biases in the measurement and the precision of the measurement.

There are three types of errors encountered in measurement systems: 1) systematic errors, 2) random errors, and 3) illegitimate errors. Systematic errors are those which result (reproducibly) from faulty calibration of equipment or from biases in the measurement. These errors must be estimated from an analysis of the experimental conditions and techniques, usually by properly calibrating the equipment. Random errors are the statistical fluctuations encountered in any measurement system. Illegitimate errors are errors which originate from mistakes or blunders in computation or measurement. These errors can be reduced or eliminated by performing the measurement correctly and avoiding careless mistakes. Sometimes the uncertainties resulting from systematic errors are estimated and then combined with the uncertainties resulting from random errors to give the overall accuracy of a measurement. A detailed treatment of the concepts of precision and accuracy is given by Bevington (1969), Campion et al., (1973), and Eisenhart et al., (1983).

1.6.2 Statistical Models

Sometimes random errors can be reduced by improving the measurement technique. If the random errors are introduced by the instrumentation, they can be reduced by using more reliable and precise instruments. If the random errors result from statistical fluctuations introduced by counting only a finite number of events, they can be reduced by counting more events.

Consider only the random error for measurements where counts are recorded. This type of measurement behaves according to a Poisson distribution (see below) where the random error associated with N counts is given by the square root of N . Thus, as the total number of counts is increased, the relative error decreases even though the absolute value of the error will be ± 10 counts or a relative error of ± 10 percent. However, if the measurement recorded 10,000 counts, the error will be ± 100 counts (i.e., 10 times the previous case) and the relative error is 1 percent (i.e., 10 times smaller than the previous case). Therefore, one should choose a counting time sufficiently long to give the number of counts with the desired relative random error. It must be realized that in addition to the random error associated with the number of counts recorded, there may be other systematic errors which must be combined with the random error to arrive at an overall error.

In a pulsed counting experiment the objective is to obtain a count rate which is proportional to say a radiation field intensity or a concentration of radioactive contamination. The count rate is usually determined by adding up the number of counts (pulses) produced by a detector due to the "source" plus background and dividing by the counting time interval. As pointed out above, the precision of the result can be improved by increasing the length of the counting interval. There is, of course, a practical limit to the time interval that is appropriate. For short-lived radioisotopes, measurements based on long counting intervals must be corrected for the decay rate of the isotope.

One of the basic concepts of statistics is the notion of a "population" with fixed properties, and the use of observations on a sample from the population to estimate these properties. Another basic concept is that random samples from the population have a predictable distribution of events.

There are four basic "distributions" of random events that are useful in nuclear radiation detection. The binomial distribution is the fundamental statistical law which governs random events and is therefore basic to detection systems (NCRP 58, 1978). The parameters which describe the distribution are shown in Table 1.3. In this type of distribution there are two results, either the event happened or it did not happen, i.e., the result of a coin toss is a head (event) or tail (not the event). This also applies to cases where there appears to be more than two choices such as rolling a die, but for any of the six faces the question can be asked for a particular value, i.e., a six was thrown or a six was not thrown.

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TABLE 1.3 STATISTICAL PARAMETERS.

POPULATION PARAMETERS	CORRESPONDING ESTIMATOR OF PARAMETER
X_i : OBTAINED VALUES X : IDEAL VALUE μ : MEAN (AVERAGE VALUE)	$(X_1, X_2, X_3, X_4, \dots, X_i)$ $= \lim_{N \rightarrow \infty} \left[\frac{1}{N} \sum X_i \right]$
$\bar{X}$: SAMPLE MEAN	$\bar{X} = \frac{1}{N} \sum X_i \sim \mu$
d_i : DEVIATION	$d_i = X_i - \bar{X}$ or $d_i = X_i - \mu$
σ^2 : VARIANCE	$\sigma^2 \equiv \lim_{N \rightarrow \infty} \left[\frac{1}{N} \sum_{i=1}^N (X_i - \mu)^2 \right]$
σ : STANDARD DEVIATION	rms VALUE OF ALL THE DEVIATIONS IN A SET

In the special case where the probability of an event occurring is small, i.e., the chance that a particular radionuclei in a sample will decay or that (in a time short compared to the half-life) a given number of nuclei will decay, the binomial distribution law can be put into a more convenient form known as a Poisson distribution. A Poisson distribution is characterized by a single parameter, $\bar{x}$, the mean, which is equal to the variance (σ^2).

The binomial distribution and its limiting case, the Poisson distribution, are both applied to discrete variables which take on integral values (e.g., the number of particles counted). However, the statistical theory of errors is usually based on the normal distribution function, which applies to a continuous variable. Therefore, the normal distribution has more generality and is also more amenable to analytical treatment. The normal probability density function $P(x)$ is a continuous function of x , and defined in such a way that the quantity $P(x)dx$ is the probability that the value of x lies between x and $x + dx$. The normal probability density function $P(x)$ is usually written as:

$$P(x) = \frac{1}{\sigma\sqrt{2\pi}} e^{-\frac{(x - \bar{x})^2}{2\sigma^2}} \quad (1.17)$$

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This is useful for describing the spread of values occurring in various types of nuclear radiation detection data. For example, in gamma-ray spectrometers, incoming radiation (monoenergetic) produces pulses with heights distributed about a mean. This function describes the major features of this distribution.

The special case of normal distribution where the variance is equal to the mean approximates the binomial and Poisson distributions for sufficiently large values of $\bar{x}$, and the further restriction that $|x - \bar{x}| \ll \bar{x}$. This relationship is illustrated in Figure 1.17 for the probability density function with $\bar{x} = \sigma^2 = 10$ for a normal distribution (continuous curve) and a Poisson distribution (histogram).

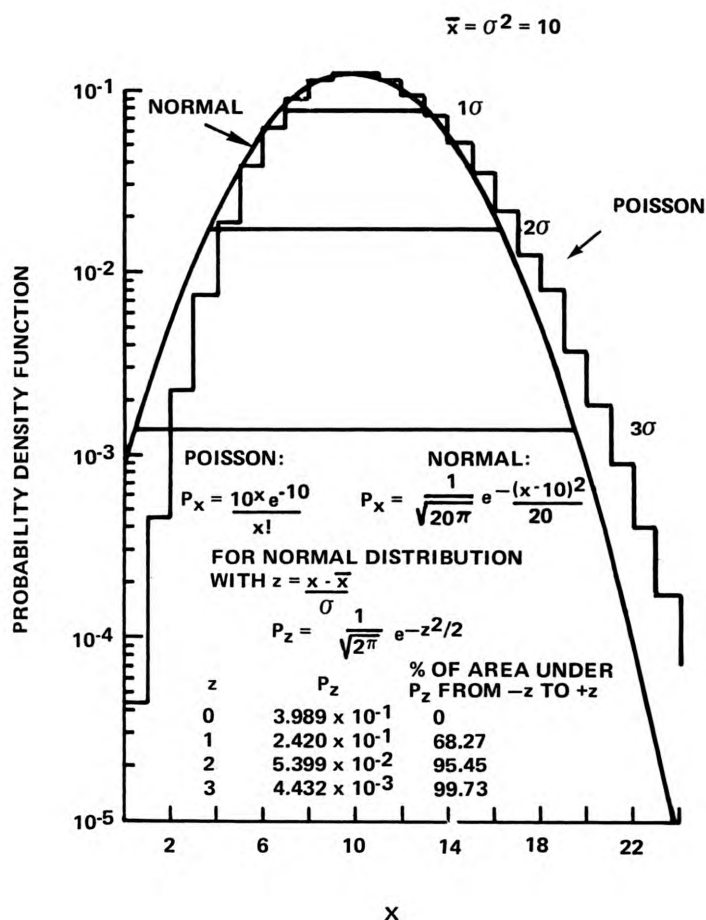


FIGURE 1.17 A COMPARISON OF THE NORMAL (GAUSSIAN) DISTRIBUTION AND THE POISSON DISTRIBUTION FOR A MEAN VALUE OF 10. THE NORMAL DISTRIBUTION WHICH IS SHOWN IS THE SPECIAL CASE IN WHICH $m = \sigma^2$.

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The results are plotted on semilog paper to emphasize the difference between these two distributions, mainly in the tails. The normal distribution is symmetrical about the mean value while the Poisson distribution is asymmetrical being lower than the normal distribution for values of x less than the mean value and greater than the normal distribution for values of x greater than the mean value.

For the normal distribution, the values corresponding to x equal to σ , 2σ , and 3σ (or $-\sigma$, -2σ , and -3σ) is also shown. Making a transformation of the variables in the normal distribution to $z = (x - \mu) / \sigma$, the value of P_z at $z = 0, 1, 2$, and 3 are listed as is the area under the density function for intervals corresponding to $\pm 1\sigma$, $\pm 2\sigma$ and $\pm 3\sigma$. For a normal distribution with a population mean, μ , and population standard deviation, σ , the intervals between the $\pm \sigma$, 2σ , etc., correspond to the corresponding confidence intervals. If the population mean and population standard deviations are only estimated by a finite number of measurements, it is necessary to use confidence intervals based on a "student-t" distribution (see example).

Another distribution function that is useful in nuclear radiation detection is the Chi-squared (χ^2) distribution. A table of percentiles of the Chi-squared distribution is available (Natrella, 1963). The value for χ^2_{α} , with $N-1$ degrees of freedom can be used as a criterion to decide if the variance computed from the N counts is larger than that expected purely due to chance. The degree of freedom is the number of independent measurements used in the estimate of the variance.

1.6.3 Error Propagation

The results that are obtained in most nuclear radiation detection experiments are not always expressed in counts or counts per unit of time. They might be expressed as activity, exposure, absorbed dose rate, etc. Thus the results of most experiments are entered into formulas which involve one or more mathematical operations to arrive at the desired quantity and unit. Consequently, the errors associated with a measurement whose numerical value (e.g., the mean value, $\bar{x}$) is used in an equation propagates through the equation producing a different error in the result. Many "propagation of error formulas" have been developed to estimate what impact an error (associated with a variable used in a mathematical operation) has on the result of that operation (Ku, 1966).

Table 1.4 illustrates some of the most useful propagation of error formulas. It is assumed that x and y are statistically independent variables and that $W = f(x,y)$, such that W is real and finite and that x and y are not zero. The formulas in Table 1.4 can be used to estimate the error in W due to uncertainties in the values of x and/or y .

As an example of some of the concepts of the last two sections, consider the calibration of a survey instrument in a radiation field

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produced with a radioactive source. The survey instrument will be placed at the calibration distance five times and a reading taken to form a replicate set of measurements. Let the readings be those given in Table 1.5. The average value is 28.4 ± 1.1 where 1.1 is the standard error of the mean. This error can also be expressed at a given confidence level using a "student-t" value (see any introductory text on statistical analysis). For this example with 4 degrees of freedom at a 99 percent confidence level, the two sided "t value" is 4.6 so the result is 28.4 ± 5.2 . If the "true" value of the reference field is not in this interval, a bias exists in the calibration procedure.

The correction factor, N, will be calculated according to the equation

$$N = \frac{\Gamma A e^{-\lambda t} / r^2}{m} \quad (1.18)$$

where Γ is the specific gamma-ray constant, A is the activity of the source in curies, λ is the source decay constant, t is the time since source calibration, r is the distance from the source to the center of the detector, and m is the meter reading.

Consider the following information for calibrating the instrument. A 100 Ci Cs-137 source with a calibration uncertainty of ± 10 percent will be used, the maximum uncertainty in placing the instrument at the calibration distance is ± 1 cm, the values of Γ and $T_{1/2}$ and their errors are $\Gamma = 0.322 \pm 0.004$ R/h/Ci at 1 meter, $T_{1/2} = 30.0 \pm 0.3$ years, systematic errors in the meter reading (i.e., due to charge collection system operation, ability to read the scale, meter linearity, etc.) is ± 10 percent, that it has been 1 year 264 days since the source was calibrated, and that instruments are calibrated at 100 cm from the source. Since the decay constant and half-life are related by

$$\lambda = \frac{0.693}{T_{1/2}}$$

the fractional error in λ will be the same as the fractional error in $T_{1/2}$ so $\lambda = 0.0231 \pm 1$ percent. Using this data, the calibration factor will be 1.09.

In addition to the random component of the error given by the standard error of the mean, there is also a systematic component. This is determined from Equation 1.18 and the formulae given in Table 1.4. The variance is given by

$$\sigma_N^2 = N^2 \left[\frac{\sigma_\Gamma^2}{\Gamma^2} + \frac{\sigma_A^2}{A^2} + 4 \frac{\sigma_r^2}{r^2} + (\lambda t e^{-\lambda t})^2 \frac{\sigma_\lambda^2}{\lambda^2} + \frac{\sigma_M^2}{M^2} \right] \quad (1.19)$$

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TABLE 1.4 ERROR PROPAGATION FORMULAS FOR SOME SIMPLE FUNCTIONS WITH STATISTICALLY INDEPENDENT VARIABLES.

FUNCTION	VARIANCE
$W(X,Y)$	$\left(\frac{\partial W}{\partial X}\right)^2 \sigma_x^2 + \left(\frac{\partial W}{\partial Y}\right)^2 \sigma_y^2$
$\frac{x}{y}$	$\left(\frac{x}{y}\right)^2 \left(\frac{\sigma_x^2}{x^2} + \frac{\sigma_y^2}{y^2}\right)$
$\frac{1}{x}$	$\frac{\sigma_x^2}{x^4}$
$\frac{x}{x+y}$	$\frac{y^2}{x^2} \left(\frac{x}{x+y}\right)^4 \left(\frac{\sigma_x^2}{x^2} + \frac{\sigma_y^2}{y^2}\right)$
$\frac{x}{1+x}$	$\frac{\sigma_x^2}{(1+x)^4}$
x^2	$4x^2 \sigma_x^2$
$\sqrt{x}$	$\frac{1}{4} \frac{\sigma_x^2}{x}$
$kx^a y^b$	$(kx^a y^b)^2 \left(a^2 \frac{\sigma_x^2}{x^2} + b^2 \frac{\sigma_y^2}{y^2}\right)$
$\ln x$	$\frac{\sigma_x^2}{x^2}$
e^{ax}	$a^2 e^{2ax} \sigma_x^2$
$\frac{\sin\left(\frac{x+y}{2}\right)}{\sin\left(\frac{x}{2}\right)}$	$\frac{1}{4} \left\{ \frac{\cos^2\left(\frac{x+y}{2}\right)}{\sin^2\left(\frac{x}{2}\right)} \sigma_y^2 + \frac{\sin^2\frac{y}{2}}{\sin^4\frac{x}{2}} \sigma_x^2 \right\}$

Since the errors specified above represent (hopefully) maximum uncertainties on the experimental parameters, they should be divided by a number like 2 or 3 to make them comparable to the random error associated with the instrument reading. This accounts for the leading factor of $1/3$ in the equation for σ_N^2/N^2 in Table 1.5. The

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TABLE 1.5 CALIBRATION EXAMPLE

TRY	READING (R/h)	$X - \bar{X}$	$(X - \bar{X})^2$
1	28.6	0.22	0.0484
2	32.7	4.32	18.6624
3	28.1	-0.28	0.0784
4	24.3	-4.08	16.6464
5	28.2	-0.18	0.0324
$\bar{X} = 28.38$		SUM = 35.4680	

$$\text{STD. DEV.} = S = \sqrt{\frac{\sum (X - \bar{X})^2}{n - 1}} = 2.98$$

$$\text{STD. ERROR OF MEAN} = \frac{S}{\sqrt{n}} = 1.13$$

$$\text{STD. ERROR OF MEAN AT 99\% CONFIDENCE} = \frac{ts}{\sqrt{n}} = (4.60) (1.13) = 5.20$$

$$N = \frac{(0.322) (100) e^{-(0.231) (1.123)}}{28.38} = \frac{30.94}{28.38} = 1.09$$

$$\begin{aligned} \frac{\sigma_N^2}{N^2} &= 1/3 \left\{ \left(\frac{0.004}{0.322} \right)^2 + (0.1)^2 + 4 \left(\frac{0.01}{1} \right)^2 + (0.0382)^2 (0.01)^2 + (0.1)^2 \right\} \\ &= 1/3 \left\{ 1.54 + 100 + 4 + 0.00 + 100 \right\} = 68.51 \times 10^{-4} \end{aligned}$$

$$\frac{s^2}{N^2} = 3 \left\{ \left(\frac{S/n}{\bar{X}} \right)^2 + \frac{\sigma_\mu^2}{\mu^2} \right\} = 3 \left\{ 15.85 + 68.51 \right\} \times 10^{-4} = 253.08 \times 10^{-4}$$

$$N = 1.09 \pm .17 = 1.09 \pm 16\%$$

fractional standard deviation for this example is then 0.083. Looking at the relative size of the error components, it can be seen that the source calibration and instrument operation terms dominate. If the calibration were done at 10 cm rather than 100 cm, the error due to source positioning would increase from 4 to 400 and thus it would dominate.

Finally, the systematic and random components should be combined and then expressed at a higher confidence level. There are many ways this can be done. A common way to obtain the overall error, s , is from the following equation:

$$\frac{s^2}{N^2} = 3 \left\{ \left(\frac{\text{Fractional}}{\text{Random Error}} \right)^2 + \left(\frac{\text{Fractional}}{\text{Systematic Error}} \right)^2 \right\} \quad (1.20)$$

The final result is

$$N = 1.09 \pm 16\% = 1.09 \pm 0.17$$

Several points should be noted in this example. First, the final error consists of random and systematic components. In this case, if the final estimation of the error were only based on the random component given by the standard error of the mean, it would be about a factor of 4 too small. Second, the systematic and random components may not be at comparable confidence levels, so some procedure must be specified to combine the results. Third, the number of significant figures in the final answer should be compatible with the combined error. As given above it is implied that one can distinguish one unit in the last significant digit. With almost a twenty percent error it would be better to express the result as $N = 1.1 \pm 0.2$.

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CHAPTER 2

CALIBRATION - AN OVERVIEW

G. T. LALOS and H. T. HEATON II

2.1 INTRODUCTION

This chapter will be concerned primarily with calibrating instruments and will only cover those aspects of source calibration which impact on instrument calibration. The object of calibrating an instrument is to quantify its operating characteristics so that its use in a given measurement situation is made in as informed a manner as possible. Calibration of an instrument is accomplished by determining its actually measured responses or readings relative to a series of known radiation values over the range of the instrument. Source calibration may be accomplished by comparing the unknown source strength relative to that of a standard. Ionizing radiation measuring instruments are generally calibrated in a known field or by a reference instrument technique. The accuracy of a measurement depends on the instrument's design, on the care taken in its calibration, and on good operating procedures such as frequent constancy tests during the time interval between calibrations, and on proper measurement techniques.

Proper instrument calibration requires that an assessment be made of its response not only to the type and energy of radiation for which the instrument was designed, but also to other radiations which may be present in practice and may contribute to the instrument reading. It also requires that mechanical and electrical integrity, and effects such as scale non-linearity, reproducibility, charge collection efficiency, pulse pile-up, charged particle equilibrium, and range-change errors be examined. Since manufacturers may not have necessary facilities for complete "type-testing" of instruments, it is desirable to check new instruments before use. A complete "type-test" will determine all the operating characteristics of the instrument with regard to radiation response, electrical characteristics and mechanical properties. A few examples of examinations which would be included in each of these three categories are for radiation: energy response, response to mixed types of radiation, location of effective center, ion current collection efficiency, and calibration or conversion factor; for electrical: reproducibility of meter movement, meter linearity, range-change characteristics, pulse pair resolution, and battery life; and for mechanical: geotropism, angular response of

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instrument, and response to impact. More detailed information on "type-testing" of instruments can be found in IEC 395 (1972), IEC 463 (1974), IEC 325 (1970), and Bramson et al. (1976). A complete "type-test" need not be repeated for all instruments built to the same design. However, before any instrument is placed into operation, the calibration factor for specified radiation(s) should be determined, the scale linearity checked, the response on various measuring ranges (range-change error) determined, and the response to radiation overload measured. The initial calibration may involve either a wide or narrow energy spectrum. If a complete energy response curve is desired, a number of monoenergetic sources may be used. The choice depends on the purpose for which the instrument is to be used.

The characteristic of an instrument which is most likely to change over a period of time is the ratio of indicated value to "true" value, i.e., the reciprocal of the calibration factor. Some instruments provide an adjustment so that the reading can be brought back to the correct value when changes occur. In general, such adjustments should be made only at a calibration laboratory having the standards needed to re-check the instrument over its whole measurement range. The change in this ratio may not be linear over the entire range of the instrument. Some instruments are fitted with a switch position for testing the battery and there may be a zero adjustment. It should be realized that the battery test may not check all the batteries. If there is doubt, the batteries should be checked with an external battery tester (voltmeter and load for battery) before calibration. The use of constancy checks is clearly important. In general, the scope of periodic recalibrations should be the same as that of the initial calibration. However, the shape of the energy response curve is unlikely to change with time for most instruments. Thus, in actual calibration procedures, instruments are often checked on each range at only a few energies.

The purpose of a particular calibration procedure will determine the conditions under which it must be carried out. A distinction should be made between rigorously controlled tests in which the physical conditions are those appropriate for "type-testing" and other tests in which the response of an instrument is evaluated under conditions similar to those in which it is to be used. The conditions of calibration should always be stated clearly in reports. When instruments are purchased, the buyer should obtain information about the nature and accuracy of any calibration that the manufacturer may have made. Such information should be a part of the specifications furnished by the manufacturer or distributor.

2.2 ORGANIZATIONAL STRUCTURE

Meaningful instrument calibration requires standard reference instruments and/or sources, adequately equipped facilities, trained personnel, and explicit procedures for relating the response of the

CALIBRATION – AN OVERVIEW

calibrated instrument to a reference instrument or source. Ideally, all calibrated instruments should be capable of having their stated accuracy traceable to national standards kept at national standards laboratories. Because it is impractical for all users in a country to interact directly with that country's national standards laboratory, a multilevel system has evolved as shown in Figure 2.1 (Eisenhower, 1982).

In the United States, the national standards are maintained by the National Bureau of Standards (NBS). The NBS interacts directly with a number of highly competent Secondary Standards laboratories in the Private, Federal, and State Sectors (Figure 2.1). For ionizing radiation, examples of possible Secondary Standards laboratories in the Federal Sector include: National Laboratories (Los Alamos, Lawrence Livermore, etc.), Army, Navy, and Air Force Laboratories, National Center for Devices and Radiological Health, etc. In the State Sector, NBS is cooperating with several states to develop pilot Secondary Standards laboratories (Neuweg, 1980). Examples of possible Secondary Standards laboratories in the Private Sector include Battelle Pacific Northwest Laboratories which has calibration facilities comparable to National Laboratories, and the set of the American Association of Physicists in Medicine's Accredited Dosimetry Calibration Laboratories for calibrating field instruments used in radiation therapy (Shalek et al., 1980). Laboratories in the "Secondary Level" calibrate instruments which may be used directly for field measurements, or for reference standards at calibration laboratories in the "Field Use Level." A directory of commercial calibration laboratories has recently been issued (NBS GCR 80-296, 1981).

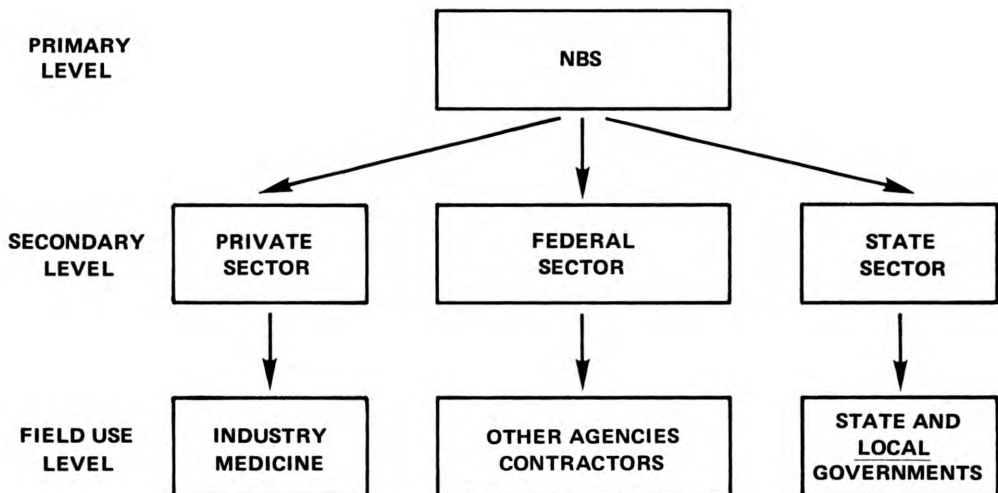


FIGURE 2.1 MULTI-LEVEL NATIONAL MEASUREMENT SUPPORT SYSTEM (EISENHOWER, 1982).

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2.3 RADIATION STANDARDS

2.3.1 Standards Hierarchy

It is possible to construct a standards hierarchy (applicable to both instrument and source calibrations) that corresponds to the organizational hierarchy shown in Figure 2.1. One of the problems with the conventional practice of calling standards "primary," "secondary," etc., is that the "primary" standard of the lower level in the hierarchy is often the "secondary" standard of the level above it in the hierarchy. The standards hierarchy given in Table 2.1 partially avoids this problem by simply labeling the standards used by organizations in the various levels as Level 1, 2, 3 or 4 standards. The table also lists uses of these standards and their associated uncertainty ranges in that level.

The uncertainty ranges given in the table include the uncertainties of all the levels higher in the chain. Thus, no standard maintained by a laboratory lower in the chain can have a total uncertainty smaller than the uncertainty assigned to the national standard for that quantity. Note, however, that with careful procedures it may be possible for a laboratory to make measurements relative to its standard with a statistical precision smaller than the overall uncertainty assigned to the standard. The overall uncertainty (systematic plus statistical) of measurements made by the calibration laboratory must be composed of a suitable combination of the uncertainty on its standard, the statistical precision of the measurement, and any biases in the measurement.

The actual values listed on the table of the uncertainties for standards for the various levels are to be taken as a guide and not as unchangeable values. In fact, there may be cases where greater accuracy is necessary for a given application than is shown in Table 2.1.

At present there is no unique set of criteria which can be applied to determine at what level in Figure 2.1 a particular calibration or measurement laboratory belongs. Criteria at the Secondary Level should at least include: an adequate quality control program to monitor internal performance, adequate documentation of procedures, participation with satisfactory results in a measurement quality assurance service provided by the National Bureau of Standards, and an overall estimate of uncertainties in the range shown in Table 2.1. Lacking these criteria, instrument and source manufacturers and commercial calibrators could fall into either Secondary or Field Level, depending on their calibration laboratory equipment and procedures. Table 2.1 reflects that at present NBS interacts directly with some users which might ultimately be in the Field Level when sufficient laboratories have been established in the Secondary Level for all the sectors.

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TABLE 2.1 STANDARDS HIERARCHY

LEVEL	USE OF STANDARD	TYPICAL UNCERTAINTY (%)	COMMENTS
1	National Standard	$\times$ 1-2 γ 1-2 β 1-2 ⁽¹⁾ 5-10 ⁽²⁾ α 1-2 ⁽¹⁾ n 1-5	Includes uncertainty on physical constants necessary to determine the quantity; represents latest state-of-the-art measurements.
2	1) Primary standard of Secondary Standards labs 2) Primary standard of organizations needing the highest level in-house standard	$\times$ 2-5 γ 2-5 β 5-15 α 2-10 n 5-15	Instrument and source manufacturers desiring calibration laboratories comparable to the Secondary Level labs in the Federal and state sectors will need the highest level standards available to them.
3	1) Primary standard for Field Level labs 2) Working standard for Secondary Standards labs	$\times$ 3-15 γ 3-15 β 10-20 α 3-20 n 7-20	These standards measured by Secondary Standards labs could serve as their working standards if they did not wish to use their primary standards for routine calibration.
4	Constancy standard	$\times$ 10-50 γ 10-50 β 15-50 α 15-50 n 20-50	The absolute value of these standards is not as important as being able to use them in a stable manner, i.e., instrument position, scattering, etc., remaining the same. It may be necessary to make corrections for source decay. These sources may be used by personnel at any level to monitor equipment performance.

(1) For radioactive sources calibrated in terms of activity or emission rate.

(2) For sources calibrated in terms of absorbed dose measured with an extrapolation chamber.

2.3.2 National Standards

In the United States the National Bureau of Standards (NBS) has the responsibility to establish, verify, maintain, and provide suitable measurement standards, and to perform calibrations and measurement quality assurance services to assure that ionizing radiation measurements made in the United States are in adequate agreement with national standards. Establishment of standards refers to the design, construction, and verification of measurement standards of a quality adequate to serve as primary national standards. Verification of these standards involves appropriate theoretical and experimental tests, and often intercomparison with comparable standards of other national and international

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laboratories. Maintenance of standards refers to periodic tests of their constancy and reliability, both by means of internal tests and by occasional intercomparison with other countries. Standards are made available by calibration and transfer of suitable instruments and sources, as needed to meet the needs of the U.S., by providing "Standard Reference Materials" (SMRs) and by offering measurement quality assurance services to test the performance of users in obtaining measurement results that are consistent with the national standards to the accuracy needed.

The remainder of this section will deal with the physical quantities of greatest interest for ionizing radiation measurements and the corresponding national standards. The physical quantities are exposure, absorbed dose, activity and fluence. The standards are free-air chambers, graphite cavity chambers, calorimeters, extrapolation chambers, radium standards and a RaBe photoneutron source.

Emphasis is placed on the special nature of national standards, i.e., on the fact that these are standards that realize the unit of a quantity from its definition. This is in contrast to all other standards lower in the hierarchy which are calibrated by comparison with the national standards either directly or through the hierarchy chain.

The Free-Air Chamber (Exposure)

Exposure is the quotient of the total charge of electrons (or positive ions) produced when all the electrons liberated by photons interacting with a small volume of air are completely stopped, and the mass of that small volume of air. The free-air chamber measures the charge liberated when photons interact in air along the beam line. A uniform electric field perpendicular to the photon beam collects the ion pairs produced. Since the photons can interact any place along the beam line, proper operation of the free-air chamber depends on charged particle equilibrium (e.g., the amount of charge entering, produced, and leaving adjacent volume elements being equal) to achieve the appropriate charge measurement. The magnitude of the unit of exposure is established by measurement of the quantities Q (total charge collected), A (cross-sectional area of the photon beam), L (length of the collecting electrode), and ρ (density of air at the time of measurement), and the determination of a number of near-unity dimensionless correction factors. Thus, the free-air chamber is not calibrated, but rather its response is established from the above quantities and from the values of certain physical constants (Loevinger, 1976). The design and construction of free-air chambers is described in detail in a NBS handbook (Wyckoff, 1957).

A set of three free-air chambers is the national measurement standard of exposure for x-rays. They cover x-ray generating potentials 10 to 60 kV, 20 to 100 kV, and 60 to 250 kV. The three NBS free-air chambers have been intercompared among themselves a number

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of times with agreement of 0.35% or better. Direct comparison with the primary standard chambers of several other national laboratories has shown agreement to better than 0.5%. Indirect comparisons of the NBS free-air chambers with other national standards have been made on several occasions at the Bureau International des Poids et Mesures (BIPM) by means of transfer standards. Again the NBS standards agreed within 0.5% with the BIPM standards and with the mean of the other national standards.

The Graphite Cavity Chamber (Exposure)

Practically, the use of the free-air chamber at atmospheric pressure is limited to photon energies less than about 300 keV. This limitation arises mainly from the fact that with increasing photon energy, the range of secondary electrons produced by the photon interactions increases, reaching a value of nearly 5 meters for the photons of Co-60. Since an atmospheric pressure free-air chamber for such photons would be far too large, high pressure free-air chambers were used at several national laboratories, and were studied at NBS (Wyckoff, 1960). An alternative approach is the use of graphite cavity ionization chambers as standards of exposure for the gamma rays of Cs-137 (0.66 MeV) and Co-60 (average: 1.25 MeV). The exposure measured from graphite cavity chambers depends on the measured volume of the cavity, the amount of charge collected, the density of air, some physical properties of the graphite and air (relative mass-energy absorption coefficients and relative mass stopping powers) and some other near unity correction factors.

The national standard for exposure for photons from Cs-137 and Co-60 is based on the averaged response of a set of six spherical chambers with active volumes from 1 cm³ to 50 cm³ (Loftus, 1969, 1974). All the chambers are made of high-purity graphite, including the central electrode.

Intercomparison of the chambers in a Co-60 beam typically shows agreement within 0.1% except for the 1-cm chamber whose response is 0.3% different from the mean response. In general, the agreement of the NBS standard graphite cavity chambers with the mean response of standard cavity chambers of other countries is the same as for free-air chambers, i.e., agreement is within about 0.5%. An uncertainty of 0.7% is assigned to the exposure graphite cavity ionization chambers (Loftus, 1974).

Graphite Calorimeters (Absorbed Dose)

Absorbed dose applies to both photon beams (x, and gamma radiation) and particle beams (beta rays, neutrons, etc.). It is defined in terms of the mean energy imparted per unit mass at the point of interest in some stated material (ICRU 33, 1980). Energy imparted to matter results in a rise in temperature, and the national standard for realization of the unit of absorbed dose in terms of its definition is a calorimeter. The special name for the unit of

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absorbed dose is gray (Gy), where $1 \text{ Gy} = 1 \text{ J/kg}$. The special unit of absorbed dose, $1 \text{ rad} = 10^{-2} \text{ Gy}$, is presently allowed, but the ICRU recommends (ICRU 33, 1980) that the SI unit be used after 1985.

NBS has two high purity graphite calorimeters for determining absorbed-dose. The first is of graphite with dimensions of $40 \times 40 \times 30$ -cm and is permanently located in one of the experimental areas of the NBS 100-MeV linear accelerator. The "portable" calorimeter is in a 15-cm diameter by 10-cm deep graphite cylinder. These two calorimeters use a "heat-loss-compensation" principle, which provides a method of measuring nearly all the heat lost from the central core to its surrounding jacket at the time of calibration (Domen, 1974).

Interpretation of the response of a calorimeter is in principle simpler than interpretation of the response of a cavity ionization chamber. On the other hand, the calorimeter is considerably more delicate and complex to build than is a cavity chamber. A comparison of their sensitivities is instructive: an absorbed dose of 3 Gy (300 rad) produces a temperature rise of about 4 mK per gram ($4 \times 10^{-3} \text{ }^{\circ}\text{C}$) in graphite; an exposure of about the same magnitude, 75 mJ/kg (300 R) liberates in air about 100 nC of charge in an ionization chamber with a volume of 1 cm^3 . Thus, if we desire to measure this magnitude of radiation with a precision of 0.1%, we must in effect be able to detect differences of $4 \mu\text{K}$ ($4 \times 10^{-6} \text{ }^{\circ}\text{C}$) per gram for the calorimeter, and about 100 pC of charge for the ionization chamber. This charge measurement is not too difficult, but such a temperature measurement requires skill, as well as complex, sensitive, and expensive equipment. Clearly, a calorimeter is simpler in principle, but considerably more complicated in practice than an ionization chamber. This, of course, is why the ionization chamber is used for routine dosimetry, and why the calorimeter is not.

The two NBS calorimeters were intercompared in 20 MeV and 50 MeV electron beams of the NBS linear accelerator, and were found to agree to within 0.1-0.2% (Domen, 1976). The NBS portable calorimeter has been compared with ionometric standards of absorbed dose in graphite phantoms, both at NBS and at BIPM, and the two methods agree to about 0.3%, which is well within the uncertainties associated with the physical constants necessary for the comparison, namely the stopping-power ratio and the mean energy expended in air per unit charge.

Extrapolation Chamber (Absorbed Dose)

Calibration of the electron beams in the 30-100 MeV energy range can be performed calorimetrically, i.e., with graphite calorimeters as discussed above. For beta sources, which result in much smaller absorbed doses than the electron beams, there is insufficient temperature change to permit use of existing calorimeters. For these sources, the national standard is an extrapolation ionization chamber.

The most versatile ionization chamber for this purpose is a

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plane-parallel plate ionization chamber in which the air gap between the collecting electrode and the polarizing electrode can be varied and the results extrapolated to zero volume. The chamber now in use at NBS is a modification made by J.S. Pruitt to an instrument described some years ago (Loevinger, 1966). The air gap in this chamber can be varied from about 0.05 mm to 20 mm. The collecting electrode can readily be changed, and a number of such electrodes are on hand. These are made of various low-atomic number materials, with collecting areas that vary from about 1 mm to 30 mm in diameter. Calibrations performed with the extrapolation chamber are reported in terms of absorbed dose by interpreting the ionization current in terms of the Bragg-Gray equation, using conventional values for the mean energy expended per unit charge, and for the stopping-power ratios.

Radium Standards

NBS has two primary radium standards. They were prepared by Prof. O. Hönigschmid in 1934, in the form of a weighed amount of a radium salt sealed in glass. They were compared with other primary Hönigschmid standards in Paris and Vienna in 1936. A number of platinum-iridium sealed radium sources were calibrated against the NBS Hönigschmid standards to serve as working standards since radium sealed in glass is not a very practical source for routine work. Intercomparison in 1955-57 between the United States, British, Canadian, and German primary standards showed these to be within about 0.2% agreement (Loftus, 1957).

Standard Photoneutron Source

The national standard photoneutron source, NBS-I, is the primary artifact standard for all fast-neutron source strength determinations and related measurements of neutron fluence rates. NBS-I is a radium-beryllium (γ, n) source containing one gram of radium. The radium, enclosed in a cylindrical platinum-iridium can 8.4 mm dia. by 8.6 mm high with 0.2 mm wall thickness, is at the center of a 40 mm dia. sphere of beryllium (Curtiss, 1949). The beryllium sphere is enclosed in a thin aluminum jacket. The emission rate has been determined, using a heavy water solution of manganous sulphate (Noyce, 1963), to be 1.257×10^6 neutrons per second with an accuracy of approximately 1.0% (1σ). Its decay rate is 0.04% per year. Intercomparison with similar neutron sources results in agreement of approximately 0.6% (1σ). A second source, called NBS-II, has been used for intercomparisons, and is identical to NBS-I except that the radium is enclosed in a 1.0 mm thick monel capsule.

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2.3.3 Transfer Standards

All standards that are below the national standards (Level 1) in the standards hierarchy (Table 2.1), are referred to as transfer standards since they transfer the standards from the higher level in the hierarchy to lower levels in the hierarchy. Note that only Level 2 standards are calibrated by direct comparison with the national standards.

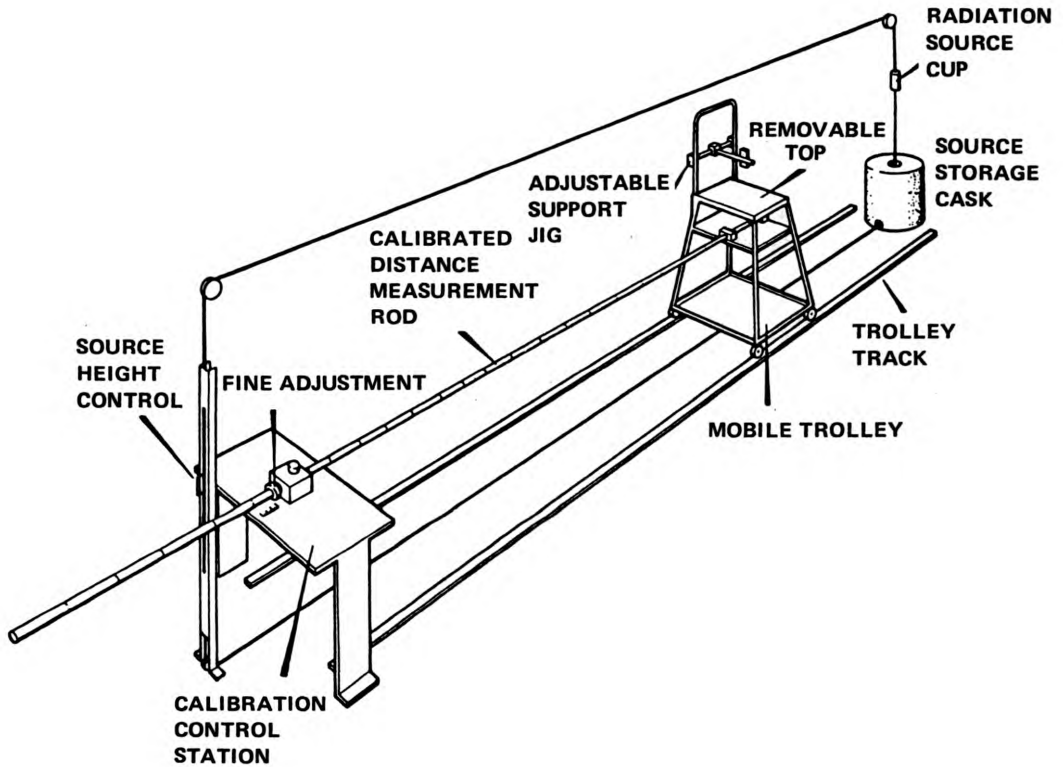


FIGURE 2.2 TYPICAL STAND AND EQUIPMENT FOR GAMMA CALIBRATION (BRAMSON, 1976).

2.4 INSTRUMENT CALIBRATION

2.4.1 Basic Calibration Techniques

There are two commonly used techniques for calibrating ionizing radiation measuring instruments. In the first technique the "unknown" instrument's response is determined in a known radiation field. In the second technique, the "unknown" instrument's response is compared against that of a calibrated reference instrument. The latter technique can be subdivided into two parts:

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1) substitution method in which the reference instrument and the unknown instrument are sequentially placed in the radiation field (this requires the field remaining constant); and 2) simultaneous method in which both the reference and unknown instrument are placed in the radiation field at the same time (this results in possible instrument cross-scattering problems). A typical calibration stand for routine gamma calibration, suitable for either technique is shown in Figure 2.2.

Known Field Technique

In this technique the radiation field is characterized from knowledge of certain parameters of the source or characterized by a known output rate determined at a specified distance. In either case, if the standard source approximates a point source, and if scattering and air attenuation are negligible, the inverse-square law can be used to calculate field strength as a function of distance from the source. Radioactive sources used for instrument calibration are generally calibrated for exposure rate at a given distance, for activity, or for emission rate (source strength). It is necessary to make an explicit correction for source decay. Machines used as known sources must have radiation monitors calibrated with a reference instrument for various machine operating conditions. Once the monitor has been calibrated, the field at the calibration point is then known in terms of the monitor response.

Reference Instrument Technique

The reference instrument technique employs either the substitution method or the simultaneous method. In calibrating instruments using the substitution method a reference instrument is placed in the radiation field and its response noted. This instrument should have been calibrated at the same energy as the radiation field in use. The instrument being calibrated is then substituted for the reference instrument (effective centers at same position) and the calibration made by comparing the two readings. In the simultaneous method the "unknown" instrument and the reference instrument are placed in the radiation field of the calibration source at the same time and the response of the two instruments noted. The two instruments should be separated sufficiently to minimize instrument cross scattering, but not so much as to result in one or both of the instruments being outside the region of uniform field strength. The reference instrument technique should always be used when the field strength of the calibration source is not accurately known.

Calibration Wells

In many practical situations, it is necessary to calibrate a large number of the same model of survey instrument. In this case one

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is interested in using a method in which the instruments can be quickly calibrated. One such method is to use a calibration well. The correct use of a calibration well depends on a slight modification of the reference instrument technique.

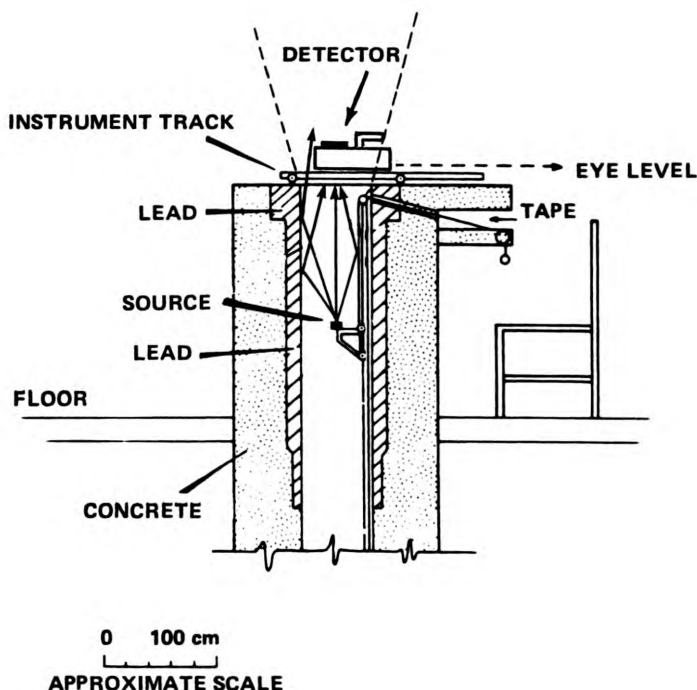


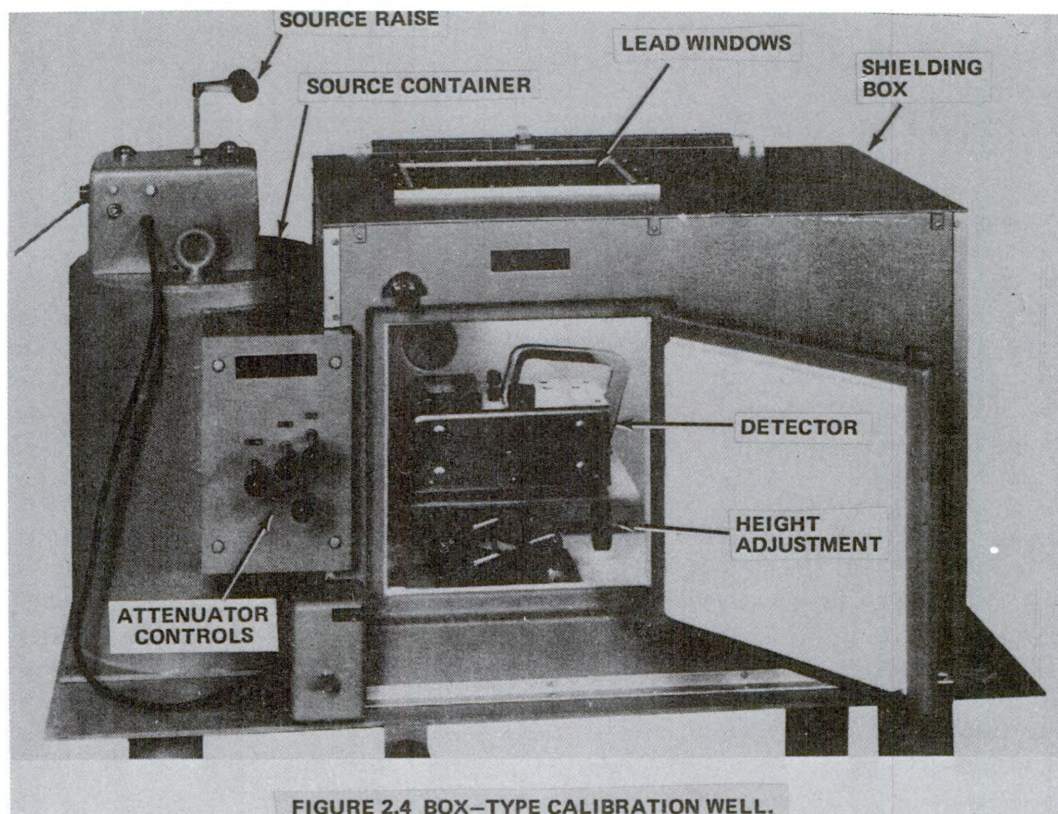
FIGURE 2.3 TUBE-TYPE CALIBRATION WELL.

To understand the reasons why the reference instrument technique must be modified, the general features of calibration wells will be reviewed. Calibration wells are usually in one of two geometries. Figure 2.3 shows a "tube" type calibration well. The intensity is varied by adjusting the height of the source in the tube. Note that radiation is scattered in the well and a portion of it can be incident on the detector. The "box" type calibration well is shown in Figure 2.4. In this case the intensity is varied by placing attenuators of various thickness in the beam line. The survey instrument is placed in a small volume so again there is scattered radiation incident on the survey instrument.

It is this scattered radiation which necessitates a modification of the reference instrument technique. This scattered radiation will have a different energy spectrum than the primary radiation. If the reference instrument has an energy dependence, the scattered radiation will result in an error in the determination of field strength. To minimize the effects due to the scattered radiation one would like to have a "reference" instrument identical to the instrument to be calibrated. In this case, the scattering of radiation both inside the instrument and in the well would be

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identical and the energy response of both the reference instrument and the survey instrument would be the same. This can be accomplished by selecting one of the survey instruments and designating it as a "reference" or pseudo-reference instrument. This "reference" instrument is then calibrated in scatter-free conditions (by either using the normal reference instrument technique or using a known field technique). The "reference" instrument is then placed in the well and its response noted. All other instruments of the same type are then placed in the well and each response compared against that of the "reference" instrument.



Two points are essential for this method to work. First, the calibration for the "reference" instrument is the scatter-free calibration. The calibration well merely provides a convenient source of radiation. Second, the reference instrument should be essentially identical to the survey instrument being calibrated.

2.4.2 Calibration Adjustments and Corrections

The correction factor determined in the calibration process is transferred to the instrument by one of the following three procedures.

Equipment Adjustment In this procedure the test instrument

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calibration controls are adjusted until the display reads correctly, i.e., reads the actual value of the radiation field (known-field technique), or gives the same reading as the reference instrument (reference instrument technique). A typical example would be the adjustment of a "Cutie Pie" ionization chamber instrument (Chapter 3) in a 100 mR/hr Cs-137 gamma radiation field to display this value.

Cross-Reference Table This procedure does not involve any instrument adjustments. The observed display readings are recorded as a function of known reference field strength (for both known field and reference instrument techniques) and this information is used to generate tables of corrected field strength (or correction factors) versus display reading. Typical examples would be pulse output from a GM counter in terms of roentgens, or channel number versus energy for gamma spectrometry.

Tolerance Certification This procedure is used mostly by calibration laboratories that handle large numbers of inexpensive but reusable personnel dosimeters. Depending on the device and its intended use, a value of the desired accuracy is chosen. If the device registers a reading outside of the chosen tolerance level it is discarded. If it indicates a dose within the chosen tolerance level it is certified as being calibrated. For example, if during a test of 100 randomly chosen TLDs (Chapter 8) two or three are outside the tolerance level they are removed from use and discarded. Thus, calibration has been performed in that the remaining TLDs will read within a given percentage of actual dose.

2.4.3 Calibration Sources and Techniques

Before an instrument is calibrated, appropriate operational checks should be made. The type of radiation field that the instrument is to be used in will influence the initial tests that are made in the instrument and possibly the energy at which it is calibrated. A description of precalibration checks is given in Section 2.6.

X- and Gamma-Ray Calibration

The calibration of photon monitoring instruments over the energy range from a few keV to a few MeV is normally performed using x-ray generators and radionuclide sources. Different field strengths should be available to accommodate the different types and ranges of instruments undergoing calibration. Field strengths may cover the range from natural background levels up to a few thousand R/hr.

At energies below 300 keV instrument calibration is normally performed in terms of exposure using an x-ray generator with a collimated beam and an ionization chamber as the reference instrument. The ionization chamber in itself should be calibrated in the next higher level in the standards hierarchy. Calibrations using x-ray generators normally employ the substitution method in order to

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minimize the necessity of corrections (scattering, etc.).

When instrument calibration is performed using an x-ray generator and the substitution method is employed one must be certain that there is no change in the x-ray beam exposure rate or spectral quality when the reference instrument is replaced by the instrument being calibrated. A monitor helps ensure that no changes have occurred (IAEA 133, 1971).

Many exposure measuring instruments have a response that is very energy-dependent, especially at photon energies below 100 keV. When studying the energy dependence of an instrument it is important that the x-ray spectrum used be as narrow as possible (Shambon, 1962). The spectrum of an x-ray beam can be substantially reduced in width and hardened (maximum intensity shifted to higher energy) by using appropriate absorbers made of high purity materials. For energy dependence measurements below 100 keV the K-fluorescence emission of different elements can be used as an almost monoenergetic source of radiation (Storm, 1965; Shambon, 1963). The improved energy definition obtained with K-fluorescence radiation over the use of heavily filtered x-ray beams is generally not necessary for radiation protection purposes. For given x-ray tube voltages, recommended filter material and radiator material can be found in ISO 4037, (1979). Alternatively, the same beam filters used in calibration at NBS (see appendix of NBS SP 250, 1982) could be adopted.

TABLE 2.2 PHOTON EMITTING RADIONUCLIDES SUITABLE FOR
USE IN INSTRUMENT CALIBRATION

RADIONUCLIDE	EFFECTIVE ENERGY (keV)	HALF-LIFE	EXPOSURE RATE CONSTANT R/(hr·Ci) at 1m
¹²⁵ I	35	60 Days	0.0044
²⁴¹ Am	60	433 Years	0.0129
⁵⁷ Co	122	270 Days	0.097
^{114m} In	192	50 Days	0.043
²⁰³ Hg	279	47 Days	0.12
⁵¹ Cr	320	28 Days	0.018
¹⁹⁸ Au	412	2.7 Days	0.23
¹³⁷ Cs	662	30.0 Years	0.323
²²⁶ Ra	830	1600 Years	0.825 ⁽¹⁾
⁶⁰ Co	1250	5.27 Years	1.30
²⁴ Na	2061	15 Hours	1.84

⁽¹⁾For radium in equilibrium with its daughter products filtered by 0.5mm of platinum.
REFERENCES: Nachtigal, 1969; Lederer, 1978; Nuclear Data, 1966-73; Kocher, 1981.

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The effective energy of an x-ray beam can be determined from the half-value layer (HVL). This is accomplished by experimentally determining the absorber thickness (usually aluminum or copper) required to reduce the beam exposure rate by 50% under scatter-free conditions. The effective energy is the energy of a monoenergetic x ray which has the same half-value layer as the x-ray beam (ICRU, 1964; Trout, 1960).

From energies of 300 keV up to a few MeV, radionuclides are generally the best choice of radiation sources for calibration. Co-60, Cs-137 and Ra-226 are most common, but other radionuclides are available, including some providing photons in the range of energies produced by x-ray tubes. Radionuclides suitable for use in instrument calibration are shown in Table 2.2. Such sources often are calibrated (ICRU 12, 1968) in terms of exposure rate at some specified distance, and proper allowance must be made for subsequent decay. Alternatively, exposure rates due to a source can be calculated from its radioactive content using the exposure rate constant. This constant includes x rays from the source. Otherwise it is the same as the previously defined (Chapter 1) specific gamma ray constant (Nachtigall, 1969). When calculating the exposure rate at a fixed distance, it may be necessary to allow for attenuation of the radiation in the source and its encapsulation.

Ideally, a radionuclide source should emit photons of a single energy and have a long half-life. In practice, however, although emission at a single energy may be the best choice for determining response, a calibration source with similar emission characteristics as the source to be monitored would be preferable for overall instrument response. The instrument and sources should be positioned well above ground on stands which introduce very little additional scatter, i.e., the stands should be constructed from a minimal amount of low-atomic number materials (Al or plastic). See Figure 2.2 for a typical calibration stand and equipment for routine gamma calibration (IAEA 133, 1971).

The radiation exposure rate used to check scale linearity in the calibration may be varied by changing the current in the x-ray tube (provided that the tube potential is kept constant), by use of nuclide sources of different activities, or by changing the distance between source and instrument. The inverse square relation may be used in many cases, but the validity of such a procedure should be established since departures from this relationship will occur when source dimensions are comparable with the source-detector distance or where scattered radiation or air attenuation is great. The validity of the inverse square assumption may be verified by measurements made with a physically small detector, thereby reducing ambiguities about the reference point of measurement.

The sensitivity of a radiation detector may vary with the angle of incidence of the radiation. One can study this effect by calibrating the instrument at various angles to the incident radiation for different energies of the radiation. Similarly, a measuring instrument designed to be worn on the surface of the body

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may differ in its response as the angle of incidence of the radiation changes.

Variations in ambient temperature and pressure will have an effect on an instrument's calibration for instruments vented to the atmosphere (e.g., air ionization chambers). In temperate climates, or under laboratory conditions, and near sea level such effects may be unimportant. If, however, the instrument is to be used at high altitudes or outdoors in very hot or cold climates, correction factors for these conditions should be made. High humidity may also have adverse effects on instrument performance, especially on insulators, or other high impedance components. Instruments with a zero-set control often enable compensation for environmental effects on electronics. If the instrument is to be used in ambient conditions which differ greatly from normal, then environmental tests should be made to assess the likely errors so that correction factors may be applied to the readings if required.

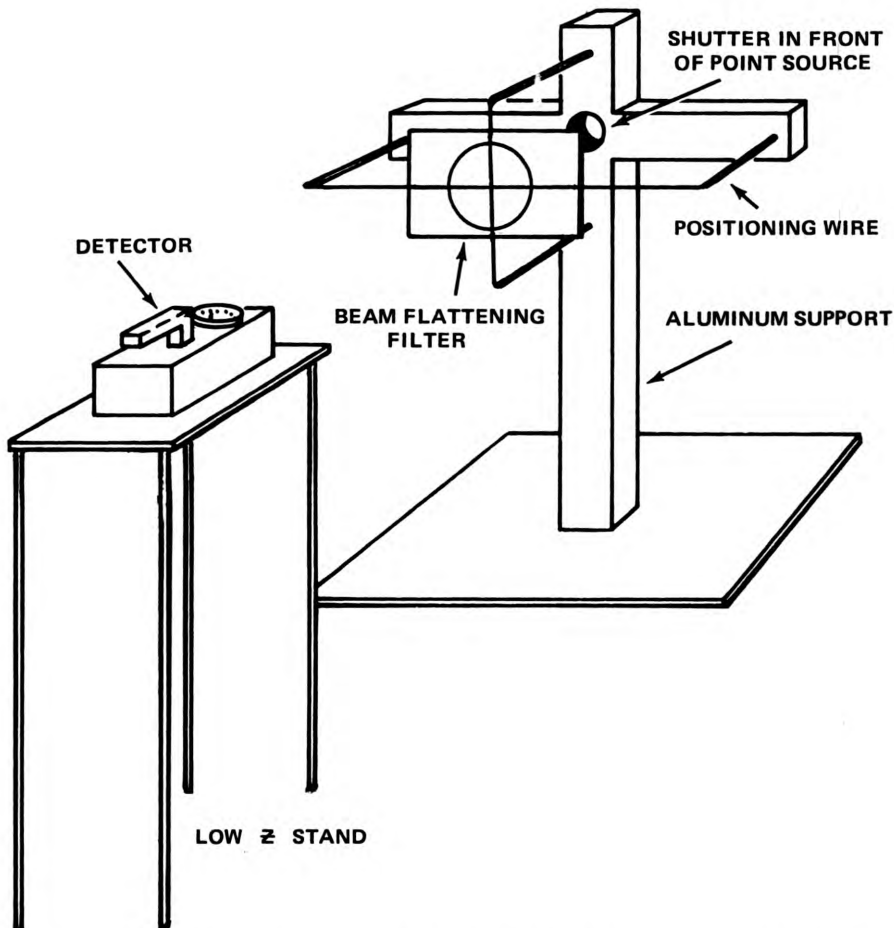


FIGURE 2.5 POINT SOURCE β RADIATION CALIBRATION FACILITY.

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Beta-Ray Calibration

Calibrating instruments for beta radiation and the interpretation of readings made with such instruments are difficult. The beta particles for a given radionuclide have a spectrum of energies up to some maximum. Hence, they are usually characterized by both an average energy and a maximum energy. For materials other than gases, the range of the beta particles is usually less than a few cm. Calibration of an instrument can be determined with a source of beta radiation which has been calibrated with an extrapolation chamber (IAEA 133, 1971; Loevinger, 1966). Instruments responding to beta particles will also respond to x and gamma radiation and to secondary electrons generated by them in air or in the instrument.

TABLE 2.3 BETA RADIATION SOURCES FOR INSTRUMENT CALIBRATION
(IN ORDER OF INCREASING $E_{\max}$)

RADIONUCLIDE ⁽¹⁾	$E_{\max}$ (MeV)	E_{avg} (MeV)	PERCENT	HALF-LIFE
^3H	0.0185	0.0057	100	12.35 Years
^{14}C	0.156	0.049	100	5.730 Years
^{35}S	0.167	0.049	100	88.0 Years
^{147}Pm	0.224	0.060	100	2.62 Years
^{45}Ca	0.257	0.077	100	164 Days
^{99}Tc	0.294	0.085	100	2.13×10^5 Years
^{60}Co	0.3179	0.0959	99.9	5.271 Years
^{185}W	0.433	0.144	100	75 Days
^{85}Kr	0.674	0.246	99	10.7 Years
^{204}Tl	0.763	0.243	98	3.78 Years
^{111}Ag	1.03	0.351	93	3.78 Days
^{210}Bi	1.16	0.394	100	5.01 Days
^{32}P	1.71	0.695	100	14.3 Days
$^{90}\text{Sr}-^{90}\text{Y}$	2.27	0.566	99	28.5 Years
^{238}U	3.26	---	---	4.49×10^9 Years
^{42}K	3.25	1.43	82	12.4 Hours

(1) Notes: Some sources accompanied by γ rays.

References: Lederer, 1978; Nuclear Data, 1966-73; Kocher, 1981.

To obtain uniform radiation fields, calibrations are commonly made with the detector window nearly in contact with a large area, flat, uniformly distributed source made of natural uranium, U-238 or Sr-90 (Cember, 1969; Gale and Peale, 1963) or at a given distance from a small source with suitable beam flatteners (Owens, 1972). Figure 2.5 shows a typical point source β radiation calibration facility. Table 2.3 includes a list of beta sources which are suitable for instrument calibration. Some of these sources also emit photons. The instrument response to these photons, together with any bremsstrahlung from surrounding materials, should be taken into consideration in the calibration. Instruments for measuring beta

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particles are usually calibrated in terms of absorbed dose to air or to tissue. The energy dependence of instruments will be less if its walls and end window are constructed of air or tissue-like material.

Alpha-Ray Calibration

Alpha sources are used for the calibration of instruments used in the field for monitoring surface contamination. Alpha-ray sources are commonly prepared by electroplating a metallic alpha-ray emitter such as uranium or plutonium onto a disc or a suitable metal such as stainless steel. The emitter may also be thinly deposited on a low Z material so as to have minimal self absorption and backscatter. The prepared source is assayed, in terms of emission rate into a known solid angle by the use of a standard counter. This is frequently either a windowless proportional counter, a proportional counter, a proportional counter with an extremely thin window, or a solid state detector. The sample itself absorbs some portion of the alpha particle energy due to its finite thickness and, where there is an appreciable path length between source and detector, geometrical straggling is also a factor. These sources themselves are usually standardized with a precision of better than 2 % by comparison with a suitable reference source standard in a counting laboratory.

Table 2.4 lists energies, abundances, and half-lives of sources suitable for calibration of alpha-ray measuring instruments. Pu-239 and natural uranium are two of the most commonly used radionuclides for calibration purposes. They are usually used as a thin metallic foil (natural uranium) or by electroplating a metallic alpha solution onto stainless steel, platinum, or other suitable metal. These preparation methods provide a uniform distribution of alpha emitter and minimize loss of activity. Evaporation of solutions is unsatisfactory for producing standard sources, for the activity tends to come away from the backing and uniform distribution of activity is virtually impossible to achieve. Because of the short range of alpha particles, standard alpha sources usually are uncovered and must be handled carefully, both from a health standpoint and to avoid loss of activity.

Inasmuch as several of the sources listed in Table 2.4 also emit photons or betas along with the alphas, instrument response to all emitted radiations must be considered when selecting a source. Ideally, the calibration source should be the same nuclide as the one the instrument is intended to measure, but this may be impractical if the instrument will be used for many (and not necessarily known) nuclides or for general alpha contamination surveys.

If only a single nuclide were to be selected as a standard source for alpha contamination survey meters, Pu-239 probably would be the choice, although safeguarding precautions (i.e., need to inventory) may be necessary. The energy of alphas emitted by this nuclide, approximately 5.1 MeV, is roughly in the middle of the range of energies of most common alpha emitters (4.1 - 6.1 MeV) and in

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TABLE 2.4 ALPHA RADIATION SOURCES FOR INSTRUMENT CALIBRATION
(IN ORDER OF INCREASING ENERGY)

RADIONUCLIDE	ALPHA ENERGY (MeV)	PERCENT	HALF-LIFE	
¹⁴⁸ Gd	3.18	100	98	Years
²³⁸ U	4.15	25	4.51×10^9	Years
	4.20	75		
	4.37	18		
²³⁵ U	4.40	57	7.1×10^8	Years
	4.58	8		
	4.717	28		
²³⁴ U	4.763	72	2.35×10^5	Years
	4.617	24		
²³⁰ Th	4.684	76	7.7×10^4	Years
	5.102	12		
²³⁹ Pu	5.143	15	2.44×10^4	Years
	5.156	73		
	5.305	100		
²¹⁰ Po	5.305	100	138.4	Days
²⁴¹ Am	5.442	13	433	Years
	5.482	86		
²³⁸ Pu	5.456	28	87.8	Years
	5.499	72		
²⁴⁴ Cm	5.764	23	17.8	Years
	5.806	77		
²⁴² Cm	6.066	26	0.44	Years
	6.110	74		
²⁵² Cf	6.076	16	2.65	Years
	6.119	84		

References: Lederer, 1978; Nuclear Data, 1966-73; Kocher, 1981.

addition it has a long half-life (24,400 yr) and readily lends itself to electroplating. Another alpha emitter that is gaining acceptance as a "single" source is Am-241 with a half-life of 433 yr and an alpha energy of about 5.5 MeV.

It is good calibration practice to employ a number of radionuclides emitting alpha particles of different energies (see Table 2.4). The range of alpha particles in materials is much shorter than the range for the same energy beta particle. Thus the ionization produced in a detector by alpha particles varies considerably with alpha particle energy, and depends on entrance window thickness, air path length, etc. The short range of an alpha particle in air (1.8 to 3.4 cm for energies listed in Table 2.4) necessitates that the instrument be calibrated and used with the detector as physically close to the radiation source as possible.

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This in turn makes the instrument response strongly dependent upon the source and detector size. Figure 2.6 shows a typical calibration setup employing a large area α radiation source.

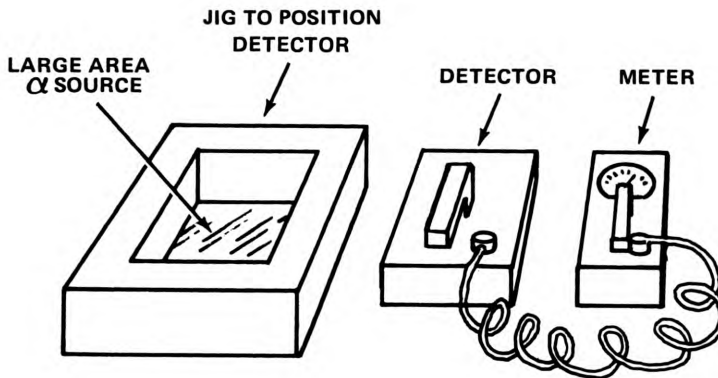


FIGURE 2.6 LARGE AREA α RADIATION CALIBRATION SETUP.

Neutron Radiation

In measuring neutron radiation, one often uses an instrument which directly measures dose equivalent. In the absence of any modifying factors, dose equivalent is the product of the quality factor, Q , and the absorbed dose at the point of interest in tissue. The quality factor relates the biological damage done by different types of radiation and depends on the linear energy transfer (LET). For photons and betas the quality factor is unity. However, for neutrons, the quality factor is a function of neutron energy and may vary between 2 and 11.

Although neutron monitoring instruments can be calibrated with neutrons produced by nuclear reactors, most neutron calibrations are made with sealed sources or accelerators. Sealed sources for calibration purposes should have conveniently long half-lives, adequate output, small physical dimensions, known neutron energy spectra, and ideally, be free from unwanted radiations. Table 2.5 lists radionuclides suitable for use in neutron instrument calibration. Anisotropy in output should be known, as should the variation of output with time. Radionuclide neutron sources are of three types: alpha-n, gamma-n, and spontaneous fission.

The alpha-n radiation sources contain an alpha emitter, such as Po-210, Pu-238, Pu-239, or Am-241 in intimate contact with a low-atomic-number element or elements, such as lithium, beryllium, boron, or fluorine, and produce neutrons distributed in energy from nearly zero to the maximum allowed by reaction kinetics. These sources can be physically small in size and are easily portable.

TABLE 2.5 SEALED RADIONUCLIDE NEUTRON SOURCES FOR INSTRUMENT CALIBRATION⁽¹⁾

SOURCE	REACTION	HALF-LIFE	AVERAGE ENERGY (MeV)	APPROX. YIELD (n/sec·Ci)	APPROX. NEUTRON DOSE RATE (mrem/hr · Ci at 1 m)	COMMENT
¹²⁴ SbBe	(Y, n)	60 Days	0.024	1.6 X 10 ⁶	0.16	High intensity IER (2)
²¹⁰ PoB	(α, n)	138 Days	2.3	6.0 X 10 ⁵	0.6	
²¹⁰ PoBe	(α, n)	138 Days	4.5	2.5 X 10 ⁶	2.9	
²²⁶ RaBe	(α, n)	1640 Years	3.9	1.5 X 10 ⁷	16.6	High intensity IER; may be yield change from ingrowth of daughters
	(Y, n)	1640 Years	0.3	1.2 X 10 ⁶	0.1	Not generally available.
²³⁷ AcBe	(α, n)	21.8 Years	4.6	~1.2 X 10 ⁶	1.4	Not generally available, high intensity IER
²²⁸ ThBe	(α, n)	1.9 Years	0.8	~2.5 X 10 ⁶	2.3	May have impurities
²³⁸ PuBe	(α, n)	87.8 Years	4.5	1.8 X 10 ⁶	2.1	May have impurities and yield change from ²⁴¹ Am ingrowth
²³⁹ PuBe	(α, n)	24,400 Years	4.5	1.5 X 10 ⁶	1.8	Also 0.6 SF n/sec—g of ²⁴¹ Am
²⁴¹ AmBe	(α, n)	458 Years	4.4	2.2 X 10 ⁶	2.6	Expensive
²⁴⁰ Pu	SF	6600 Years	FISSION	3500	3.7 X 10 ⁻³	Expensive, high intensity IER
²⁴⁴ Cm	SF	18.4 Years	FISSION	1.5 X 10 ⁵	0.2	Expensive, high yield, useful point source, high intensity IER
²⁵² Cf	SF	2.65 Years	FISSION	4.3 X 10 ⁹	4.6 X 10 ³	

(1) From Kathren, 1975.

(2) Ionizing Electromagnetic Radiation.

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They can be fabricated with neutron yields up to the $10^6 - 10^8$ neutron per second range, and have relatively low-intensity accompanying photon emission. Recommended alpha-n radiation sources are based on Pu-238, and Am-241. Pu-239 is also an attractive source but its use may be complicated by an ingrowth problem (ICRU 10b, 1964).

The gamma-n radiation sources consist of a gamma emitter of suitably high photon energy, such as Sb-124 or Ra-226, placed in close proximity to a low-atomic-number element, usually deuterium or beryllium. Reacting gamma rays produce neutrons by photonuclear reactions. Since most of the photons produce no neutrons, these sources also have intense photon fields. This may create personnel exposure problems, as well as interference with instrument response. Each gamma photon produces a unique energy neutron. For Ra the spectrum of gamma photons results in a spectrum of neutron energies. For Sb, most of the neutrons above the gamma-n threshold in Be are at a single energy. This results in about 95% of the neutrons having an energy of 23 keV (Harrison, 1978). The short half-life of Sb-124 (60 days) and the low neutron energy are important considerations for the use of this type source. Ra-226 gamma-n radiation sources are more generally useful, are physically small in size, and can be obtained with neutron yields up to the $10^8 - 10^9$ neutron per second range.

Spontaneous fission neutrons, emitted in one branch of the decay of Cf-252, have a fission-type neutron spectrum in the energy range from approximately thermal to 15 MeV. These sources can closely approximate an idealized point source, can be obtained in a wide range of source strengths, i.e. to $10^6 - 10^{11}$ neutrons per second, and have a known spectrum of accompanying photons (ICRU 26, 1977). The relatively short half-life (2.65 yr) is a limitation to long-term use.

Radionuclide neutron sources can be sent to the National Bureau of Standards for calibration in terms of emission rate. When this is not possible, the neutron source emission rate can be determined by the long counter or manganese bath technique. On the other hand, since particle accelerators and nuclear reactors cannot be sent out for calibration, they must be standardized for instrument calibration by use of standard neutron instruments or techniques. In addition, the neutron output of these facilities must be monitored continuously during instrument calibrations. Standard neutron instruments and techniques include the precision long counter, associated particle counters (for certain reactions with certain accelerators), nuclear emulsions, fission foils, activation foils, and the manganese sulfate bath. A more detailed description of calibration and characteristics of neutron sources may be found in ICRU 10b (1964), ICRU 13 (1969), and ICRU 26 (1977).

The most widely used method for calibrating neutron sources for emission rate is with a manganese bath. This method depends on having a calibrated neutron source and making relative measurements. The source is placed at the center of a large, often spherical, tank which contains a saturated solution of MnSO_4 . The solution acts as a

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moderator and the slow neutrons produced are captured by the stable Mn-55 to give Mn-56 which decays with a half-life of 2.578 hr. The dimensions of the bath are chosen so that the escape of neutrons from the bath is kept as low as possible. Samples of the solution are measured and an absolute determination of the Mn-56 activity is made. This requires knowing the absolute efficiency for gamma rays from Mn-56 for the detectors used for this measurement. The detectors most commonly used are NaI crystals. A larger number of corrections have to be made in the estimation of the source emission rates. Neutrons may undergo fast reactions in the sulphur and oxygen if their energies are above the thresholds for these reactions; other neutrons are scattered back from the solution and reabsorbed in the source. A number of other correction factors also have to be applied and these are discussed, along with a more detailed description of the technique, in Axton (1961), De Juren (1955), and Geiger (1959).

Particle accelerators produce intense neutron fields by the interaction of accelerated charged particles, such as protons, deuterons, or tritons, on low-atomic-number target materials such as deuterium, tritium, and lithium. Available field strengths exceed the maximum emission rates of radionuclide sources thus circumventing the difficulty with nuclide sources of calibrating instruments on their higher ranges while still maintaining reasonable source-to-detector separation. Important characteristics of these neutron fields are variable intensity up to very high values (yields in excess of 10^{12} neutrons per second in some cases), occurrence of radiation in a brief pulse, monoenergetic neutron emission for any given beam-target-detector angular relationship, and lack of portability. Neutron output is a complex function of accelerator and target parameters and may be expected to vary with time even though measured machine parameters remain constant. Therefore, neutron output must be monitored constantly during instrument calibration work, and standard instruments or techniques must be used to establish neutron field values. The neutron fluence can be determined with a long counter. The efficiency of the long counter is energy dependent and when used with an accelerator, its efficiency can be determined with a Cf-252 source of known emission rate. For more details, see the use of a long counter in Section 2.4.5 and particularly Table 2.13.

2.4.4 Types of Sources and Reference Instruments

This section will present a brief overview of the relationship between the various classes of instruments discussed in this document and the calibration techniques of Section 2.4.1. The tables which follow are merely intended to be a guide with more detailed information given in the appropriate chapter. For any particular calibration method given in the tables, a different calibration method may be more appropriate for a given application of an instrument.

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Before listing the common calibration methods for the various instrument classes, it will be useful to summarize the types of radiation commonly measured by the various instrument classes. The common types of radiation measured by instrument class are indicated in Table 2.6. A numerical code is used in Table 2.6 to indicate the likelihood that a given class of instrument will be used for the measurement of a particular type of ionizing radiation. This code is: 1) frequently, 2) infrequently, and 3) almost never.

TABLE 2.6 TYPES OF RADIATION COMMONLY MEASURED BY
VARIOUS INSTRUMENT CLASSES.

RADIATION	ION. CHAMBER	PROP. COUNTER	GM COUNTER	SCINTILLATOR	SEMICONDUCTOR	TLD	FILM	ACTIVATION FOIL	CHEM. DOSIMETER	CALORIMETER	CERENKOV
x	1	2	1	1	1	1	1	3	1	1	3
γ	1	2	1	1	1	1	1	2	1	1	3
β	1	1	1	1	1	1	1	3	1	2	1
α	1	1	1	1	1	3	3	3	3	3	3
n	1	1	3	1	1	1	1	1	3	2	3

Legend: 1—Frequently, 2— Infrequently, 3 — Almost Never

Table 2.7 summarizes the common radiation sources for the "known field" technique for calibration of the various classes of instruments. If the output of a machine is calibrated with a reference instrument and then monitored with a suitable instrument, it can then be considered to be a "known field." Only the "substitution method" of the "reference instrument" technique is considered, since in the "simultaneous method" the output from the machine does not need to be known because both the reference instrument and instrument to be calibrated are in the same field at the same time.

Table 2.8 lists the types of instruments employed in the "reference instrument" technique and the types of radiation with which they are commonly used. There is not a complete overlap with Table 2.6 since, in practice, reference instruments are not commonly used with all the types of radiation for which a given class is suitable for measuring. Also, not all instruments listed may be suitable if very accurate results are desired, i.e., what may be suitable as a reference instrument for a 10% calibration may not be suitable for a 3% calibration.

The "pseudo-reference" technique discussed in the section on calibration wells can be extended to any case in which one is

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calibrating a large number of the same model of instrument, by selecting one of these for calibration in free space, and then using it as a "reference" instrument in some particular configuration and comparing other instruments of the same model against it. This is designated as "Type" in Table 2.8.

TABLE 2.7 RADIATION SOURCE TYPES COMMONLY USED WHEN CALIBRATING INSTRUMENTS WITH THE KNOWN FIELD TECHNIQUE.

<u>RADIONUCLIDE SOURCE</u>		ION. CHAMBER	PROP. COUNTER	GM COUNTER	SCINTILLATOR	SEMICONDUCTOR	TLD	FILM	ACTIVATION FOIL	CHEM. DOSIMETER	CALORIMETER	CERENKOV
γ	(¹³⁷ Cs, ⁶⁰ Co, ²²⁶ Ra) EXPOSURE	1	2	1	1	1	1	1	3	1	3	3
β	(³ H, ¹⁴ C, ⁸⁵ Kr) ACTIVITY, ABSORBED DOSE	1	1	1	1	1	1	1	3	1	3	1
α	(²³⁹ Pu, natU, ²⁴¹ Am) ACTIVITY	1	1	1	1	1	3	3	3	3	3	3
n	(γ , n; α , n; S.F.) EMISSION RATE	1	1	3	1	1	1	1	1	3	3	3
<u>MACHINE SOURCE (*)</u>												
x	RAYs (X-RAY GENERATOR)	1	2	1	1	1	1	1	3	1	2	3
γ	(VAN De GRAAFF, LINAC)	1	2	1	1	1	1	1	3	1	2	3
e	(VAN De GRAAFF, LINAC)	1	2	1	1	1	1	1	3	1	1	1
α	(VAN De GRAAFF, CYCLOTRON)	1	1	1	1	1	3	3	3	3	3	3
n	(d, T GEN., VAN De GRAAFF, LINAC, CYCLOTRON)	1	1	3	1	1	1	1	1	3	3	3

(*) For fields from machines with monitors calibrated against reference instruments thereby giving fields known in terms of calibrated monitor output.

Legend: 1—Frequently, 2—Infrequently, 3—Almost Never

2.4.5 Instrument Calibration Techniques

Instruments can be classified into two use categories: those used mainly to make measurements in the field, and those used mainly to make measurements in a laboratory. Each of these two categories could be further classified into instruments measuring dosimetric quantities, and those measuring radioactivity. Examples of typical instruments in these four categories are:

Field - Dosimetry: Survey instruments measuring exposure.*

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Field - Radioactivity: Alpha Contamination Meter.

Laboratory - Dosimetry: Reference ionization chamber measuring exposure.

Laboratory - Radioactivity: $4\pi\beta\gamma$ ionization chamber.

TABLE 2.8 REFERENCE INSTRUMENTS COMMONLY USED WHEN CALIBRATING INSTRUMENTS WITH THE REFERENCE INSTRUMENT TECHNIQUE

REFERENCE INSTRUMENT	RADIATION TYPE				
	γ	x	β or e	α	n
ION CHAMBER EXPOSURE, ABSORBED DOSE	1	1	2	3	1
EXTRAPOLATION CHAMBER ABSORBED DOSE	3	2	1	3	3
RE-ENTRANT TYPE ION CHAMBER ACTIVITY	1	1	1	2	3
$4\pi\beta$ PROPORTIONAL COUNTER ACTIVITY	1	1	1	3	3
CALORIMETER ABSORBED DOSE	1	1	1	3	3
TLD	1	1	1	3	1
FILM	1	1	1	3	1
CHEMICAL DOSIMETER	1	1	1	3	3
LONG COUNTER NEUTRON FLUENCE	3	3	3	3	1
NEUTRON ACTIVATION FOIL	3	3	3	3	1
"TYPE" *	1	1	1	1	1
GM COUNTER	}				
SCINTILLATION DET.					
SEMICONDUCTOR DET.					

* See text for discussion.

Legend: 1 - Frequently, 2 - Infrequently, 3 - Almost Never

This section will focus mainly on calibration of instruments (both field and laboratory) for making dosimetric measurements and will include some discussion on calibration of field instruments for measuring (or detecting) radioactivity of samples. It will not cover calibration of laboratory instruments used to measure radioactivity. This last topic is as broad in scope and is covered very well in a NCRP Report (NCRP 58, 1958).

As seen in Section 2.4.1, there are two basic techniques for calibrating instruments: in a known field or against a reference instrument. Both of these techniques imply that at a given time and at a given point in space, the value of the radiation field in which the instrument to be calibrated is placed is known or can be

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calculated. In the following sub-sections the method of determining the value of the reference field at a given time and point will be given explicitly for both the "known field" and "reference instrument" techniques.

Once one knows the value of the reference field at a particular point in space and at a particular time, one can place any type of instrument at that point and compare its observed reading against the reference value. The correction factor (CF) for the instrument being calibrated will be just the ratio of the reference value to the observed value,

$$CF = \text{reference value/observed instrument reading} \quad (2.1)$$

If the units associated with the reference field value are not the same as the instrument being calibrated, this is a conversion factor rather than a simple scale correction factor. Since it does not make any difference what type of instrument is being calibrated in the reference field, the general principles of performing calibrations will be covered in this chapter, while any specific fine points of calibrations for a particular type of instrument will be covered in the chapter for that instrument.

Correction Factors for Determining the Reference Field

When instruments and sources are calibrated, it is for a specific set of conditions. If the conditions are different when these instruments and sources are used, it will be necessary to correct for each perturbing factor by applying an appropriate correction factor, N , to match the calibration conditions. This applies to all instruments in the calibration hierarchy given in Table 2.1. It is necessary for personnel using an instrument at one level in the hierarchy to correct the observed instrument reading so that it corresponds to the conditions at the time it was calibrated by personnel at the next higher level in the hierarchy.

Let $f_R(t_0, r_0)$ be the correction factor for either a reference instrument or calibrated source output at a unit distance as determined by a laboratory in one level of the hierarchy. Then for the reference instrument technique, the reference field, $F_R(t, r)$, at time t , and distance r , at a calibration laboratory in the next lower level of the hierarchy is

$$F_R(t, r) = f_R(t_0, r_0) R_{\text{obs}} \prod N_i \quad (2.2a)$$

where R_{obs} is the observed reading of the reference instrument and $\prod N_i$ is the product of all the correction factors for perturbations. The corresponding relationship from the known field of a radioactive source is:

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$$F_R(t, r) = \frac{f_R(t_0, r_0)}{r^2} \prod N_i \quad (2.2b)$$

In addition to applying correction factors to determine the reference field at the calibration laboratory, it may also be necessary to apply correction factors to the observed readings of the instrument being calibrated, M_{obs} , to relate its reading to a specified set of conditions, M_{ref} .

$$M_{ref} = M_{obs} \prod N_i \quad (2.3)$$

In the material which follows, the correction factors for various perturbing factors will be considered for both calibration techniques. In actual practice some of the correction factors may be sufficiently close to unity so that they need not be applied to achieve the desired level of accuracy in the calibration.

Reference Instrument Technique

Reference Instrument Conversion (or Correction) Factor: The reference chamber should be calibrated at NBS or at one of the Secondary Standards Laboratories discussed in Section 2.2. The conversion factor will be for a specific radiation characterized by some energy related parameter(s), e.g., beam energy, half-value layer, machine constant potential, etc.; and a particular value of the radiation field. The conversion factor for the reference chamber will be designated as N_R . Ignoring all other corrections, the exposure rate of the reference field, $\dot{X}_R$, will be

$$\dot{X}_R = N_R \left(\frac{R_{obs}}{t} \right) \quad (2.4)$$

where R_{obs} is the observed integrated response of the reference instrument for time t . This is analogous to Equation 2.2a with $F_R(t, r) = \dot{X}$, $f_R(t_0, r_0) = N_R$ and R_{obs} replaced by R_{obs}/t . Note: In this chapter, a dot over a variable indicates the time rate of the variable, e.g., $\dot{x} = x/t = \text{exposure rate}$.

1) **Pressure - temperature correction:** This correction must be made for all gas filled chambers which are vented to the atmosphere and is necessary because for a given chamber volume, the density of air (and thus the mass of air within the given volume) will change with pressure and temperature. The calibration report for the reference instrument will state at what temperature, in $^{\circ}\text{C}$, and pressure (P_R and T_R) the calibration factor was determined. If the pressure and temperature in the laboratory at the time the reference

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instrument is used are P_L and T_L respectively, the calibration factor for the reference instrument must be multiplied by the correction factor N_{PT} where

$$N_{PT} = \frac{(273.2 + T_L) P_R}{(273.2 + T_R) P_L} \quad (2.5)$$

The reference temperature is usually one of the following: 0, 20 or 22°C. The reference pressure is almost always 760 mm of Hg (=29.921 inches of Hg = 101.325 kPa = 1013.25 mbars).

2) Energy dependence: Most reference chambers exhibit some energy dependence. If the conditions at the calibration laboratory do not exactly match those of the laboratory calibrating the reference instrument, an explicit correction, N_E , should be made. This can best be done by having the laboratory calibrating the reference instrument calibrate it at energies above and below those used in the calibration laboratory and interpolating to the calibration laboratory conditions.

3) Saturation current: This correction applies to reference ionization chambers used in the current (as opposed to pulse) mode. The current from the ionization chamber depends on the voltage at which the ionization chamber is operated and on the value of the radiation field in which the chamber is placed (Boag, 1966). These values should be stated on the calibration report of the reference ionization chamber. If the chamber is to be used at either a lower voltage or in a more intense field, the chamber should be checked to determine if all the ion current is being collected. This can be done by placing the ionization chamber in a constant field and measuring the observed current as a function of applied voltage. For continuous radiation fields, the inverse of the ion current is proportional to the inverse square of the applied voltage. A fit of these values (Figure 2.7) will give the inverse of the saturation current, i_0 .

For pulsed radiation fields, the ion current is more nearly proportional to the inverse of the voltage. Boag has described a two-voltage analysis for determining the collection efficiency for this case (Boag, 1980).

Let N_S be the correction factor for lack of complete ion current collection.

4) Electrometer corrections: In most cases the output from the reference chamber is measured by a separate instrument. Current output from a reference chamber is usually measured with an electrometer. The electrometer can be operated in either a "current" mode (e.g., measuring exposure rate) or in a "charge" mode (e.g., measuring total exposure). In either case the linearity of a particular range should be checked and a correction factor for each range should be determined. Both the current and charge mode can be checked with an appropriate combination of precision current sources, standard capacitors, precision voltage sources and accurate

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timers. Let N_L be a correction factor for the scale linearity and N_{ER} be a correction factor for the electrometer range.

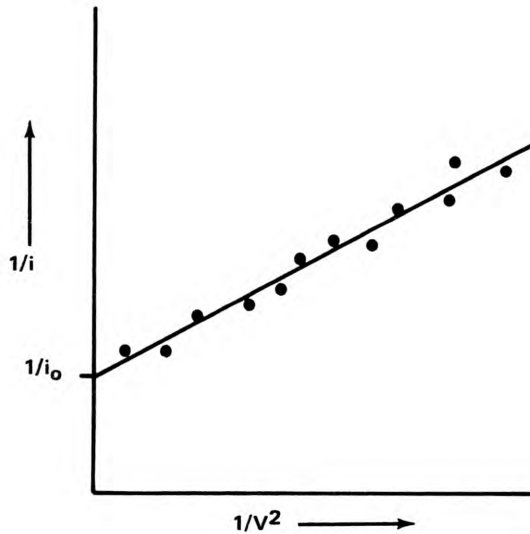


FIGURE 2.7 PLOT OF OBSERVED ION CURRENT VERSUS APPLIED VOLTAGE TO DETERMINE THE SATURATION CURRENT.

5) Dead time correction: If the pulse output from the reference chamber is recorded on a counting system, corrections will have to be made for pulses separated by less than the resolving time of the system. This can be done with pulsers and oscilloscopes or with radioactive sources (see NCRP Report 58).

6) Leakage current: When measuring small currents any leakage of charge should be accounted for. This charge leakage could be associated with the reference chamber or the electrometer and both should be checked. The leakage current will change when the voltage to the reference chamber is changed so the leakage current should be measured each time the voltage is applied to the reference chamber if it is known that this current could cause unacceptable errors in the final reading. It is possible for leakage current to increase due to the presence of radiation. This should be checked for by measuring the leakage current immediately before and after the radiation field has been removed. It will be assumed that the electrometer leakage current is negligible and that the measured leakage current from the reference instrument includes ionization due to naturally occurring radiation. The correction factor due to this will be N_{BKG} .

7) Conversion from exposure to absorbed dose: If the reference instrument is calibrated in terms of exposure and an accurate measure of absorbed dose to either air or tissue at some depth is desired, it will be necessary to make many additional corrections. These corrections are necessary for instruments used for radiation therapy but are unnecessary for instruments used for

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radiation protection when the approximation that the exposure (in R) equals the absorbed dose (in rads) equals the dose equivalent (in rem). For further information on accurate absorbed dose measurements see Johns and Cunningham (1974), ICRU 14 (1969), ICRU 17 (1970), ICRU 21 (1974), ICRU 23 (1973), ICRU 24 (1976), and NCRP 69 (1981).

Known Field Technique

Source Output Conversion Factor: The output of a radioactive source can be reported in several different ways:

Exposure rate at a given distance, $\dot{X}_0$
 Activity, A
 Emission rate (source strength), $\dot{S}_0$
 Kerma rate, $\dot{K}_0$
 Absorbed dose rate, $\dot{D}_0$
 Fluence rate, $\dot{\Phi}_0$
 Dose equivalent rate, $\dot{H}_0$

For the simple case where no corrections are necessary, the field at a distance r from the source for these various source output characterization is:

$$\dot{\Phi} = \frac{(\dot{S}_0/4\pi)}{r^2} \quad (2.6a)$$

where $\dot{S}_0$ is the emission rate (or source strength) into 4π steradians.

$$\dot{X} = \frac{(\dot{X}_0 r_0^2)}{r^2} \quad (2.6b)$$

The reference point, r_0 , is often taken to be at unit distance.

The exposure rate can also be expressed in terms of source strength, $\dot{S}_0$, for a particular energy photon, and activity, A. Let f_i be the fraction of decays resulting in the emission of a gamma ray with energy E_i . Then

$$A = \dot{S}_0 / f_i$$

and

$$\dot{X} = \frac{A}{r^2} \frac{e}{4\pi W} \sum E_i \left(\frac{\mu_{en}}{\rho} \right)_i f_i$$

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where μ_{en}/ρ is the mass energy absorption coefficient in air at E_i , and W/e is the average energy to produce an ion pair in air per electric charge (33.85 J/C = 33.85 eV/ion pair, ICRU 31, 1979). The term in the bracket is the conversion factor from activity to exposure rate, i.e., the exposure rate constant

$$\dot{X} = \frac{(\Gamma A)}{r^2} \quad (2.6c)$$

In a similar way

$$\text{Absorbed dose rate:} \quad \dot{D} = \frac{C_D A}{r^2} \quad (2.6d)$$

$$\text{Kerma rate:} \quad \dot{K} = \frac{C_K A}{r^2} \quad (2.6e)$$

$$\text{Dose equivalent rate:} \quad \dot{H} = \frac{C_H A}{r^2} \quad (2.6f)$$

where C represents the conversion factor from activity to the desired dosimetric quantity. Let N_{SO} represent the appropriate expression for source output characterization, i.e., the expression in parentheses in Equation 2.6. This equation is then analogous to Equation 2.2b with $F_R(t, r) = \dot{\Phi}, \dot{X}, \dot{D}, \dot{K},$ or $\dot{H}$, and $f_R(t_0, r_0) = N_{SO}$.

1) Source decay: All sources must be corrected for decay since the time they were calibrated. The correction factor for source decay, N_D , is

$$N_D = e^{-\lambda t} \quad (2.7)$$

where t is the time since calibration, and the decay constant (λ) is related to the half-life ($T_{1/2}$) given in Tables 2.1 - 2.4 by

$$\lambda = \frac{\ln 2}{T_{1/2}} \quad (2.8)$$

To determine the decay correction, $T_{1/2}$ and t should be in the same time units, i.e., both in seconds, or both in years, etc.

2) Source attenuation: There are two source attenuation corrections which may have to be applied depending on how the source was calibrated. These are corrections for attenuation within the source itself and attenuation within the source encapsulation. If

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the source output is reported in terms of exposure rate at some distance, these factors have been included in the measurement of the output and no correction is necessary. If the source calibration measured the source output in the same way as above and then used some exposure rate constant (not necessarily those in Table 2.2) to convert to a source activity, again no correction is needed. In this case one should use the same value of the exposure rate constant as the source manufacturer and not the current best value listed in Table 2.2. If the source output is truly activity, then it will be necessary to make a correction for attenuation within the source and its encapsulation. The correction factor for attenuation in a thickness t , and density ρ , N_{SC} , is

$$N_{SC} = e^{-\left(\frac{\mu}{\rho}\right)_c (\rho t)_c} \quad (2.9)$$

where c refers to the capsule material. For gamma rays μ/ρ can be taken to be the total attenuation coefficient (Hubbell, 1969) and for neutrons

$$\left(\frac{\mu}{\rho}\right)_c = \frac{N_o \sigma_c}{M_A} \quad (2.10)$$

where N_o = Avogadro's constant (6.022×10^{23}), and σ_c = total neutron cross section (see ICRU 26, 1977; BNL-325; Schwartz, 1974) of the encapsulation material, and M_A is the molar mass. If the encapsulation is so thick that multiple scattering results in some of the radiation originally scattered out of the direction of the detector rescattering into the direction of the detector, it is necessary to multiply the value of N_{SC} by a suitable buildup factor (Rockwell, 1956; Goldstein, 1959).

The situation for source attenuation is more complicated since the decay can occur anywhere in the source and the radiation can travel in any direction. This means that one will have to choose some representative distance, $\bar{x}$, as typical of the radiation pathlength in the source. Then the correction factor for attenuation within the source itself, N_{SS} , is

$$N_{SS} = e^{-\left(\frac{\mu}{\rho}\right)_s (\rho \bar{x})_s} \quad (2.11)$$

where s refers to the source material. For thick sources it may also be necessary to include a buildup factor. The selection of $\bar{x}$ becomes more critical as source dimensions increase since some of the radiation may become completely absorbed within the source.

If there is a spectrum of radiation from the source, the

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correction factors should be averaged over the spectrum of the radiation.

If the source output is given in terms of emission rate, the necessity of including source attenuation corrections will depend on how the emission rate was measured and reported on the calibration certificate.

3) Air attenuation: The basic form of the air attenuation correction factor, N_A , is

$$N_A = e^{-\left(\frac{\mu}{\rho}\right)_a (\rho r)_a} \quad (2.12)$$

The attenuation coefficient for air, $(\mu/\rho)_a$, is determined in the same manner as for source encapsulation.

The total attenuation coefficients for photons in air are given in Table 2.9 (Hubbell, 1982):

TABLE 2.9 TOTAL ATTENUATION COEFFICIENT FOR AIR

E (MeV)	μ/ρ (cm ² /g)
0.661 (Cs)	0.07715
0.800 (Ra)	0.07074
1.25 (Co)	0.05681

The density of air relative to STP is

$$\rho = 0.001293 \frac{273.2}{(273.2 + T)} \left(\frac{P}{760}\right) \frac{q}{\text{cm}^3} \quad (2.13)$$

where P is the pressure (in mm Hg) at the time the source is used and T is the corresponding air temperature (in °C). Note that this density correction is just the inverse of N_{PT} since N_{PT} applies to exposure or dose which has the air mass, or density, in the denominator.

If a finite air path is part of the measurement determining the source output, then r represents the distance from the calibration point to the effective center of the detector, otherwise it is the distance from the edge of the source to the edge of the detector.

4) Source anisotropy: The emission rate from a source may not be isotropic due to non-spherical source construction and difference between in-scatter and out-scatter in the source encapsulation. This can be either measured, or in some cases calculated. Designate this correction factor by N_{NI} .

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Both Techniques

1) Timing end-errors: Due to the way an x-ray shutter operates or the way a source is positioned, the actual irradiation time may not correspond to the preset time interval. This timing end-error can be determined as follows (see NCRP Report 69). Make a long exposure for time t_1 and let the corresponding meter reading be M_1 . Divide the interval t into n shorter segments. Make n exposures of total time $t_2 = (t_1/n)$ and let the total reading for these n exposures be M_2 . Then the timing error has occurred once in the reading M_1 and n times in the reading M_2 . Let Δt be the timing error where

$$\Delta t = \frac{M_2 t_1 - M_1 t_2}{n M_1 - M_2} \quad (2.14)$$

If Δt is positive, the actual irradiation time is found by adding Δt to the preset time, whereas if Δt is negative the actual irradiation time is found by subtracting Δt from the preset time. Let the correction factor due to timing end-errors be N_T .

TABLE 2.10 NET INCREASE IN RESPONSE DUE TO NEUTRONS SCATTERING IN AIR.

	INCREASE PER METER (IPM)	
	BARE ^{252}Cf	MODERATED ^{252}Cf
	(%)	(%)
FLUENCE	1.2	4.0
DOSE EQUIVALENT	1.0	1.5
NTA FILM, POLYCARBONATE TRACK ETCH DOSEMETER	0.5	0.9
9" SPHERICAL REMMETER	1.0	2.3
ALBEDO DOSEMETER	1.1	3.0
3" SPHERE	1.7	4.5

2) Air scatter: If the air path is sufficiently great that multiple scattering can occur, it is necessary to multiply N_A by a buildup factor (or its equivalent). For reasonable path lengths the buildup factor for x and gamma rays for air is taken to be unity. For neutrons, scatter by air is more important than absorption by air. Source neutrons travelling directly toward the detector can be scattered so that they never reach the detector (out-scatter) and conversely, source neutrons not travelling toward the detector can be scattered into the detector (in-scatter). Let N_{AAS} denote the difference between in-scattered and out-scattered neutrons. This difference has been calculated (Schwartz and Eisenhauer, 1982) for a few selected cases for neutrons from bare ^{252}Cf and ^{252}Cf in a D_2O sphere. The results are reported in percent increase per meter in in-scattering over out-scattering. This means that (detector)

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response should be decreased by the percentages given in Table 2.10 to obtain the free field responses, i.e., $N_A N_{AS} = 1 - (IPM/100)r$ where r is the source-detector distance.

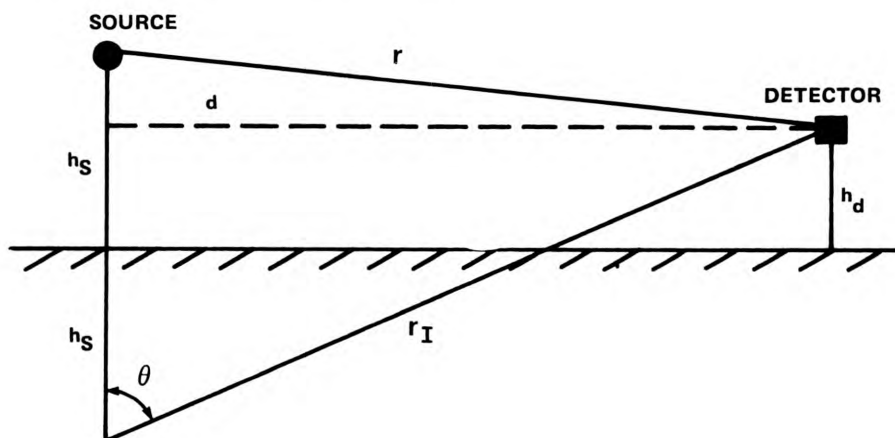


FIGURE 2.8 COORDINATES FOR CALCULATING SINGLE SURFACE RETURN.

3) Room return: Radiation can be scattered from large area surfaces such as floors into detectors. In an analogy to electrostatic electricity, Eisenhauer (1965) has developed an image source technique for calculating reflections of both gamma rays and neutrons from a single surface. The geometry is shown in Figure 2.8. Here r is the source to detector distance, r_I is the distance from the image source to the detector, h_s and h_d are the distances of the source and detector from the reflecting surface and d is the perpendicular distance between the source-image source line and detector. Let F_s and F_d respectively be the field due to the reflected (scattered) radiation and direct radiation, then

$$F_s/F_d = \left(\frac{r}{r_I}\right)^2 R(\theta, \frac{h_d}{h_s}) \quad (2.15)$$

where the reflection coefficient R (which is a function of θ and h_d/h_s) for a ^{60}Co gamma photon scattering from concrete has been calculated by Monte Carlo techniques and is plotted in the reference. The reflection coefficient increases rapidly from about 0.03 for $\cos \theta = 0$ to a maximum of 0.12 at $\cos \theta = 0.22$ and slowly decreases to 0.08 at $\cos \theta = 1$. Experimental data on the reflection coefficient are given for fast neutrons scattering from water and concrete and calculations are made for several types of one-velocity scatter. Schwartz and Eisenhauer (1982) expressed Equation 2.15 as

$$F_s/F_d = 2\alpha g \left(\frac{\sigma}{\sigma_r}\right) \left(\frac{r}{r_I}\right)^2 \cos \theta \quad (2.16)$$

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where α is the albedo of the reflecting surface (0.54 for episcadmium neutrons for concrete decreasing by about 20% for saturated soil), g is a factor to account for anisotropic detector response, σ_r and σ the spectrum averaged response for the reflected and direct neutrons respectively, $\theta = \tan^{-1} d/(h_s + h_d)$, and $r^2_I = (h_s + h_d)^2 + d^2$.

Values of $g(\sigma_r/\sigma)$ for a bare and moderated Cf source are given (Schwartz and Eisenhauer, 1982) in Table 2.11.

TABLE 2.11 CALCULATED VALUES OF THE FACTOR $g(\sigma_r/\sigma)$ FOR SINGLE SURFACE REFLECTION.

	BARE ²⁵² Cf	MODERATED ²⁵² Cf
FLUENCE	1.0	1.0
DOSE EQUIVALENT	0.37	0.6
NTA FILM, POLYCARBONATE	0.2	0.3
TRACK ETCH DOSEMETER		
9" SPHERICAL REMMETER	0.68	0.75
ALBEDO DOSEMETER	1.0	0.6
3" SPHERE	1.8	1.1

In an enclosed concrete room, each neutron makes about 2 1/2 traversals before being captured. These room-scattered neutrons are essentially uniformly distributed throughout the room. In analogy to Equation 2.16, the ratio of instrument response to reflected and direct radiation for a bare Cf-252 source in a concrete room with $r \ll h$, is

$$F_s/F_d = 5.6 g \left(\frac{\sigma_r}{\sigma} \right) \left(\frac{r}{r_c} \right)^2 \quad (2.17)$$

with $4\pi r_c^2 = \sum A_i$ where A_i is the area of the i th surface of the room and the summation is over the six room surfaces.

The reference field at the calibration point due to both direct and scattered components is

$$F_R(t, r) = \frac{f_R(t_0, r_0)}{r^2} + F_s$$

Combining this equation and Equation 2.17

$$F_R(t, r) = \frac{(1 + S r^2) f_R(t_0, r_0)}{r^2} \quad (2.18)$$

where $S = 5.6 g(\sigma_r / \sigma_0) (4\pi / \sum A_i)$. The room return correction

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factor is

$$N_{RR} = (1 + Sr^2) \quad (2.19)$$

If the room return correction factor is applied to the instrument reading rather than to the reference field, the reciprocal of Equation 2.19 should be used. For those cases where $g(\sigma_r/\sigma)$ is not known, S can be determined from the slope of the line obtained by plotting $M_{obs} r^2$ vs r^2 for various r .

For scattering from a single surface, Jenkins (1980) has used the Monte Carlo code MORSE to study neutrons from a PuBe source reflected from concrete and gives formulae which do not explicitly contain the albedo of the reflecting surface, and which fitted the scattered component to within 30%. The expression for fluence is

$$\Phi = \frac{S_0}{4\pi r^2} \left[1 + \frac{1.52 (r_I/r)}{(1 + 0.1E)(1 + (r_I/r)^3)} \right] \quad (2.20)$$

where S_0 is the source emission, E is the neutron energy in MeV, and r_I is the same as in Equation 2.17. The expression for dose equivalent is

$$H = \frac{S_0 C_\Phi}{4\pi r^2} \left[1 + \frac{0.75 (r_I/r)}{1 + (r_I/r)^3} \right] \quad (2.21)$$

where C_Φ is the spectrum averaged conversion factor from fluence to dose equivalent.

McCall (1978, 1979) has also used the computer code MORSE to calculate the scattered component of accelerator produced neutrons in concrete rooms. He found the scattered neutrons were constant in the room and that the scattered neutron fluence was given by

$$\Phi_s = k_1 \frac{S_0}{S} \quad (2.22)$$

where S_0 is the fast neutron source strength and S is the area of the room. The constant k_1 depends on the energy of the neutron spectrum and was 4.6 for tungsten-shielded medical linear accelerators and 5.4 for lead-shielded medical linear accelerators. For fluence measuring instruments $S_0 = 4\pi r^2 \Phi_0$ and $g(\sigma_r/\sigma) = 1$, so Equations 2.17 and 2.22 are the same providing $k_1 = 5.6$. McCall also found that the average energy of the scattered neutrons was 0.24 times the average energy of the primary neutrons.

4) Shadow shield: The corrections for air scatter and room return can be determined by the shadow-shield method. The principles are shown in Figure 2.9.

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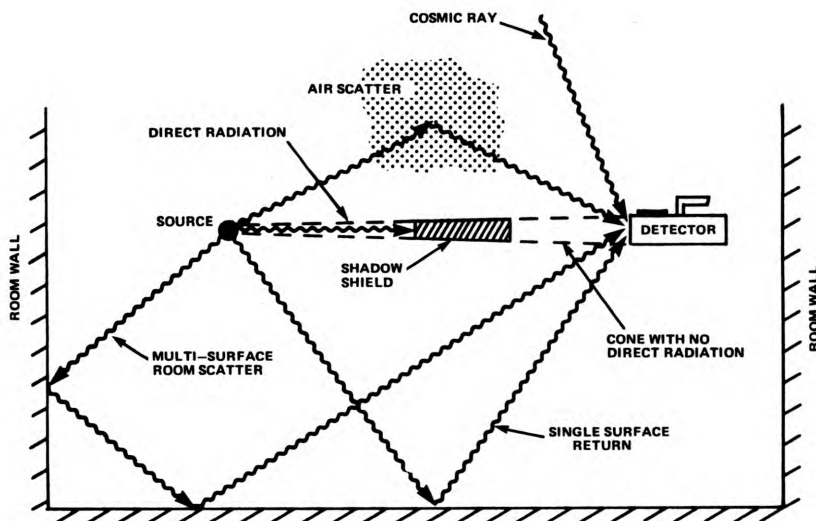


FIGURE 2.9 RADIATION REACHING DETECTOR IN
SHADOW SHIELD MEASUREMENT.

First, a measurement is made with the source unexposed (or shutter closed) and no shadow shield to measure the response due to background plus leakage, R_{BKG} . Second, a measurement is made with the source exposed (or shutter open) which measures the response due to the source, background, leakage, room return, and air scatter. Finally, a measurement is made with the shadow shield in place to measure the response due to background, leakage, room return, and air scatter.

The shadow shield should shield just the detector from the direct response of the source and should not be so large that it significantly changes the air scatter component reading of the detector or significantly "shadows" the back wall.

The above correction factors are summarized in Table 2.12. Measurements may show that some of the corrections are near enough to unity that they do not need to be applied for the desired accuracy of the calibration.

Consider the reference instrument technique. Knowing the value of the reference field at a point in space at a particular time allows us to place the effective center of any instrument at that point and compare its response with the known value to determine the correction factor for the instrument being calibrated, i.e., the observed response of the instrument is being compared with the known value of the field existing when the instrument being calibrated is not present. In the following assume the reference field is known in terms of exposure. Let $(R_s)_c$ be the response of the instrument being calibrated (c) to the reference field. Then the correction factor will be

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$$CF = \frac{\dot{X}_{Ref}}{(R_S/t)_C} \quad (2.23)$$

where t is the time for instruments which integrate over the total exposure, or

$$CF = \frac{\dot{X}_{Ref}}{\langle \dot{R} \rangle_C} \quad (2.24)$$

where $\langle \dot{R} \rangle_C$ is the average reading over the exposure time for those instruments which read rate.

TABLE 2.12 SUMMARY OF CORRECTION OR CONVERSION FACTORS FOR REFERENCE INSTRUMENT AND KNOWN FIELD TECHNIQUES.

TECHNIQUE	SYMBOL	CORRECTION OR CONVERSION FACTOR
REFERENCE-INSTRUMENT	N_R	CONVERSION FACTOR FOR REF. INSTRUMENT
	N_{PT}	PRESSURE-TEMPERATURE CORRECTION
	N_E	ENERGY DEPENDENCE OF REF. INSTRUMENT
	N_S	SATURATION CURRENT EFFECT
	N_{BKG}	LEAKAGE CURRENT + NATURAL BACKGROUND
	N_{DT}	DEAD TIME CORRECTION (PULSE INSTR.)
	N_L	ELECTROMETER SCALE LINEARITY
	N_{ER}	ELECTROMETER RANGE CORRECTION
	N_T	SHUTTER TIMING END-EFFECT CORRECTION
	N_{RR}	ROOM RETURN
	N_{AS}	AIR SCATTER
KNOWN FIELD	N_{SO}	SOURCE OUTPUT CONVERSION
	N_D	SOURCE DECAY
	N_{NI}	SOURCE ANISOTROPY
	N_{SA}	SOURCE ATTENUATION
	N_A	AIR ATTENUATION
	N_{AS}	AIR SCATTER
	N_{RR}	ROOM RETURN
	N_T	SHUTTER TIMING END-EFFECT CORRECTION

The observed response of the instrument being calibrated will consist of the following components

$$(R_{obs})_C = (R_{Source} + R_{Air\ Scatter} + R_{Room\ Return} + R_{Bkg} + R_{Timing\ end\ offset})_C$$

The response due solely to direct radiation will be

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$$R_s = (R_{obs})_c N(c)$$

where

$$N(c) = (N_{AS} N_{RR} N_{BKG} N_T)_c \quad (2.25)$$

and corrects for the other effects measured by the instrument being calibrated. If we want the source response referenced to some pressure and temperature, $N(c)$ must be multiplied by $(N_{PT})_c$.

Similarly, the observed response of the reference instrument to the source will be

$$(R_{obs})_R = (R_{Source} + R_{Air \text{ Scatter}} + R_{Room \text{ Return}} + R_{Bkg} + R_{Timing \text{ end} \text{ offset}})_R$$

and the response to the direct component of the reference field will be

$$(R_s)_R = (R_{obs})_R N(R)$$

where

$$N(R) = (N_{AS} N_{RR} N_{BKG} N_T)_R \quad (2.26)$$

and similarly corrects for effects measured by the reference instrument.

The value of the reference field for the reference pressure and temperature will be

$$\dot{\chi}_R = N_R \left(\frac{N_{PT} R_s}{t} \right)_R N(R) N_{ER} N_L \quad (2.27)$$

and the overall correction factor for the instrument being calibrated is

$$CF = \frac{\text{ref. value}}{\text{obs. reading}} = N_R \frac{(R_{obs}/t)_R}{(R_{obs}/t)_c} N_{CF}^R \quad (2.28)$$

where

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$$N_{CF}^R = \frac{(N_{PT}^R N_{AS}^R N_{RR}^R N_{BKG}^R N_T^R)_R}{(N_{PT}^R N_{AS}^R N_{RR}^R N_{BKG}^R N_T^R)_C} N_{ER}^R N_L^R N_E^R N_S^R \quad (2.29)$$

A correction term has been included to account for the possibility of the energy used in the calibration being different than the energy at which the reference chamber was calibrated and for saturation current effects.

For a good electrometer, $N_L N_{ER}$ is approximately unity, and assuming the reference instrument and test instrument are used in beams of the same energy, $N_E = 1$, and the terms in the ratio can be measured. To first order this ratio, N_{CF}^R , is unity.

Now consider the known field technique. The reference field is given by

$$\dot{X}_R = \frac{N_{SO}}{r^2} N(S) \quad (2.30)$$

where

$$N(S) = N_D N_{SA} N_A N_{AS}$$

and N_{SO} represents the calibrated source output in the desired dosimetric quantity. The correction factor for the instrument being calibrated is

$$CF = \frac{\text{ref. value}}{\text{obs. reading}} = \frac{N_{SO}}{r^2} \left(\frac{1}{R_{\text{obs}}/t} \right)_C N_{CF}^S \quad (2.31)$$

where

$$N_{CF}^S = \frac{N_D N_{SA} N_A N_{AS}}{(N_{PT}^R N_{RR}^R N_{BKG}^R N_T^R)_C} \quad (2.32)$$

This correction factor is not necessarily near unity as was the case in the reference instrument technique. The known field technique depends on knowing the value of r , whereas the reference instrument technique only requires that the effective centers of both instruments be at the same spot.

X-Ray Instrument Calibration

X-ray instruments are normally calibrated with an x-ray machine, although they can also be calibrated with an Am-241 source (60 keV). There are two types of calibrations of interest: 1)

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energy response of the instrument, and 2) overall correction factor of the instrument. If one is only interested in the energy response of the instrument one normally uses fluorescence x-rays or heavily filtered x-rays (see ISO 4037), whereas if one is interested in a calibration factor one generally calibrates the instrument with a spectrum which matches the field measurement as nearly as possible.

Reference Instrument Technique

When using x-ray machines, the reference instrument technique is the most common method. In calibrating x-ray instruments it is necessary to characterize the x-ray beam in which they are calibrated. This is generally done by specifying the amount of added filter materials; the kilovoltage of the x-ray unit (preferably constant potential - less than 10% ripple (ISO 4037); the half-value layer (HVL); the homogeneity coefficient (HC), the ratio of the first HVL of Al to the second HVL of Al; or the effective energy.

If the half-value layer of an x-ray machine is measured with an energy dependent detector, it will be necessary to correct the observed readings as a function of added material since the HVL of the new beam consisting of the original added filter material plus the added material of the HVL measurement changes each time material is added. The HVL also depends on the source - detector distance since changing this distance will change the spectrum due to absorption in air.

The measured exposure rate has a power law variation with the kilovoltage of the x-ray machine, $\dot{X} = aV^n$ where n is usually between two and three. The measured exposure rate is directly proportional to the tube current, $\dot{X} = a + bI$ where a is due to dark current and possibly due to amount of ripple present when the tube current changes. To account for variations in the reference field between exposures with the reference chamber present and with an instrument to be calibrated present, a transmission monitor can be placed in the x-ray beam line.

When using a reference instrument, the dependence on the source-effective center distance, r , is small. The main concern is to ensure that the effective center of both the reference instrument and the instrument to be calibrated are at the same place. There is another r dependence, namely, beam uniformity. Let the dimension of the detector perpendicular to the x-ray beam axis be $2x$. Then the fluence at the edge of the detector is reduced from that of the center of the detector by a factor of $r^2/(r^2 + x^2)$. ANSI N325 (1978) recommends that r be greater than 7 times the detector dimension. In addition to beam uniformity depending on r , there is an additional dependence on self-absorption effects of the electron beam in the anode of the x-ray tube. This is called the "heel effect." See ICRU 10b (1962) for more discussion of this effect.

Typical instruments used as reference chambers for calibrating x-ray instruments are Shonka-Wyckoff or "R Chamber" ionization chambers. The reference x-ray field, $\dot{X}_R$, measured with one of these

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reference instruments per monitor count, M_R , is

$$\frac{\dot{X}_R}{M_R} = \frac{N_R}{M} \left(\frac{N_{PT}^{R_{OBS}}}{t} \right)_R N(R) \quad (2.33)$$

where R_{OBS} is the electrometer reading, t is the preset time, M is the monitor reading with the reference instrument present, and $N(R)$ is given by Equation 2.26.

Let $(R_{OBS})_c$ be the reading of any type of x-ray measuring instrument placed in this field (e.g., TLD, GM tube, solid state detector, etc.) and let CF be the correction factor by which we need to multiply the observed reading by to obtain the "true" reading. Then the corrected test instrument reading per monitor count must equal the reference field per monitor count, M_c , or the correction factor is

$$CF = \frac{M_c}{M_R} N_R \frac{(R_{obs}/t)_R}{(R_{obs}/t)_c} N_{CF}^R \quad (2.34)$$

where N_{CF}^R is given by Equation 2.29.

Known Field Technique

As already mentioned the known field technique is much less common with x rays. The most common source is Am-241 which emits 60 keV gamma rays. In addition to gamma rays, there are also alpha rays which can interact with oxygen and nitrogen to produce neutrons by the (α, n) reaction. The basic principles of using an Am-241 source are the same as for a gamma source discussed in the next section.

Gamma-Ray Instrument Calibration

Both the reference instrument technique and the known field technique are commonly used to calibrate gamma-ray instruments. The most common sources are Cs-137 and Co-60, and to a lesser extent Ra. If one wants higher energy gamma rays, a common method is to use a charged particle reaction with an accelerator.

Reference Instrument Technique

The basic principles of using a reference instrument for gamma fields is essentially the same as for x-rays and the basic method can be found there. For gamma rays one usually specifies the effective energy which in this case is simply the average energy of the gamma rays. With gamma-ray sources, monitors are not normally used, but if charged particle reactions with accelerators are used, monitors should be used.

The one major difference between using a reference instrument

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for x rays or gamma rays is if the gamma-ray source is in a calibration well. As seen in Section 2.4.1, calibration in a calibration well is a two step process in which for each type of instrument calibrated an instrument of that type is selected to serve as a pseudo-reference instrument. The instrument is first calibrated in free field conditions by determining its free field correction factor, CF_{PR} , either using the known field technique or the reference instrument technique. Let $\dot{X}_R$ be the free reference field value, let $R_{PR}(F)$ be the reading of the pseudo-reference instrument in the free field, and let $N_{PT}^{PR}(F)$ be the pressure-temperature correction for the pseudo-reference instrument in the free field. The free field calibration factor is

$$CF_{PR} = \frac{\dot{X}_R}{N_{PT}^{PR}(F) R_{PR}(F) / t} \quad (2.35)$$

The pseudo-reference instrument is now placed in the calibration well and its reading is taken in the well, $R_{PR}(W)$, with a corresponding pressure-temperature correction, $N_{PT}^{PR}(W)$. Finally, the instrument being calibrated (of the same type as the pseudo-reference instrument) is placed in the well. Depending on which of the calibration adjustment procedures of Section 2.4.2 is being used, one can determine the correction factor of the instrument being calibrated in two ways. One can adjust this instrument to read the same as the pseudo-reference instrument so that the correction factor for both is the same and equal to the free field calibration. Alternatively, one can form a calibration table by comparing the pressure-temperature corrected readings of the two instruments. In this case the correction factor for the instrument being calibrated, CF_T , will be related to the free field calibration factor of the pseudo-reference instrument by

$$CF_T = CF_{PR} \frac{R_{PR}(W)}{R_T(W)} \frac{N_{PT}^{PR}(W)}{N_{PT}^T(W)} \quad (2.36)$$

Known Field Technique

When one thinks of sources it usually is of point sources. The case for irradiators containing gamma sources is essentially the same but one needs to correct for an additional scattering component for scattering from the surfaces of the irradiator. These corrections are usually small enough not to require the use of a pseudo-reference instrument described in the calibration well case.

As indicated in Section 2.5.1, the output of the source (reference field) can be expressed in terms of exposure rate at a

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given distance, activity, or emission rate.

1) Reference field, $\dot{X}_R$, when output is measured in terms of exposure rate, $\dot{X}_O$, at a reference distance, r_O , is

$$\dot{X}_R = \frac{(\dot{X}_O r_O^2)}{r^2} N(S) \quad (2.37a)$$

where r = distance from the effective center of the source to the effective center of the instrument being calibrated, and

$$N(S) = N_D N_A N_{SA}$$

where N_A = air attenuation past the point at which the source was calibrated. The reference distance is often taken to be at a unit distance so r_O often does not appear in this type of equation.

2) Reference field, $\dot{X}_R$, when output is expressed in terms of activity, A , is

$$\dot{X}_R = \frac{(\Gamma A)}{r^2} N(S) \quad (2.37b)$$

where Γ is the exposure rate constant. If the source calibrator explicitly measured the exposure at some distance and then converted to an effective source activity,

$$N(S) = N_D N'_A N_{SA}$$

where N'_A = air attenuation past the point at which the source was calibrated, and if the actual activity of the source is determined

$$N(S) = N_D N_A N_{AS} N_{SA}$$

where N_A is air attenuation for entire path.

3) Reference field, $\dot{X}_R$, when output is given in terms of emission rate of photons with a particular energy. In this case it is necessary to know the decay scheme to be able to determine both the activity and exposure rate constant.

Let $(R_{OBS})_c$ be the observed reading of the test instrument placed in the reference field. The correction factor for all three cases will be the same

$$CF = \frac{\dot{X}_R}{(N_{PT} R_{OBS}/t)_c N(c)} = \frac{N_{SO}}{r^2} \frac{1}{(R_{OBS}/t)_c} \frac{N(S)}{N(c)} \quad (2.38)$$

where $N(c)$ is given by Equation 2.25, N_{SO} is the appropriate

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expression for source output, and $N(S)$ is the corresponding correction factor as discussed above.

In addition to dependence directly on r^2 , there is an additional dependence on r through the non-uniformity of the beam over the detector area in the same manner as discussed for x-ray fields.

Neutron Instrument Calibration

When dealing with neutrons, it is convenient to discuss three energy ranges, thermal, moderate, and fast. The boundary between moderate and fast is not sharp as is the case for thermal neutrons.

Reference Instrument Technique

Thermal Neutrons

1) BF_3 or ^3He counter: As an example of this kind of detector consider a BF_3 counter. This counter is operated in the proportional region and detects alpha particles by the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction. By operating in a proportional mode, good separation between the alpha particles produced by the (n,α) reaction and electrons produced by the (γ,e) reaction can be achieved.

These detectors operate in a pulse mode where the individual events are recorded separately rather than in the current mode as are the reference ionization chambers used in x-ray and gamma-ray instrument calibration. This means that a correction for dead time may be necessary, but there is no need for electrometer corrections. Since BF_3 counters are sealed chambers, no pressure-temperature correction is necessary.

The reference instrument is usually calibrated in terms of fluence (or fluence rate) incident on the detector. The fluence rate in the reference field at a particular point and time in space will be

$$\dot{\Phi}_R = N_R \left(\frac{R_{\text{obs}}}{t} \right)_R N'(R) \quad (2.39)$$

where N'_R is the conversion factor in terms of fluence rate per detector pulse, and the correction factor $N'(R)$ is

$$N'(R) = (N_{\text{DT}} N_{\text{AS}} N_{\text{RR}} N_{\text{BKG}} N_{\text{T}})_R \quad (2.40)$$

2) Gold foils: These can be used in two ways: either by using the constants of the thermal capture reaction on gold, or by irradiating gold foil in a known fluence, and use this to calibrate the counting system.

The activity induced in a gold foil, A , will be

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$$A = n \sigma_{\text{capt.}} \times a \dot{\Phi} \quad (2.41)$$

where n is the number of atoms/cm³ = N_0/M_A (N_0 = Avogadro's constant and M_A is the molar mass, σ_{CAPT} is the gold capture cross section (98.65 ± 0.09 barns), x is the foil thickness, a is the foil area, and $\dot{\Phi}$ is the fluence rate.

For long irradiation it will be necessary to correct for decay during the irradiation. For irradiation time t , the activity at the end of irradiation, $A(t)$, is

$$A(t) = A(1 - e^{-\lambda t}) \quad (2.42)$$

For Au-198, $T_{1/2} = 2.696$ days.

If one can absolutely count the activity induced in the gold foil, the fluence rate of the thermal neutron beam can be determined.

The other method of using gold foils is to irradiate them in a known fluence and then use them to calibrate the counting system. Let $C_{\Phi A}$ be the conversion factor from fluence rate to activity ($=n\sigma x a$). For a gold foil with known activity, the observed count rate (corrected for dead time) in the experimental counting system using the standard is

$$\dot{C}_S = \epsilon A_{\text{STD}} N_D^S = \epsilon C_{\Phi A} \dot{\Phi}_{\text{STD}} N_D^S \quad (2.43)$$

where ϵ is the counter efficiency, $\dot{\Phi}_{\text{STD}}$ is the fluence rate used to irradiate the standard, and N_D^S is the decay correction from the time the standard was originally irradiated to the time it was counted, i.e., $e^{-\lambda t}$. The count rate due to foil irradiation in the users reference thermal beam will be

$$\dot{C}_R = \epsilon A_R N_D^R = \dot{C}_S \frac{N_D^R}{N_D^S} \frac{A_R}{A_S} = \dot{C}_S \frac{N_D^R}{N_D^S} \left(\frac{C_{\Phi A}}{A_S} \right) \dot{\Phi}_R \quad (2.44)$$

and the reference field fluence rate will be

$$\dot{\Phi}_R = \frac{N_D^S}{N_D^R} \frac{\dot{C}_R}{\dot{C}_S} \frac{A_S}{C_{\Phi A}} = \frac{\dot{C}_R}{\dot{C}_S} \frac{N_D^S}{N_D^R} \dot{\Phi}_{\text{STD}} \quad (2.45)$$

If irradiation times are long and counting times are long, explicit corrections for these effects must be made.

3) Fission Chambers: These are ionization chambers (or

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proportional counters) which detect the ionization produced when the fission fragment travels through the counting gas.

Fission chambers can also be used in two modes. The simpler is to have the output calibrated in terms of a known thermal fluence. In this case the same expression for the reference field is obtained as for the BF_3 counter case (Equation 2.39). The other case is if the mass per unit area of the fission deposit material is known. This case is very similar to the gold foil case. The count rate in the fission chamber due to the reference field will be

$$\dot{C}_R = \frac{mN_0}{M_A} \langle \sigma_{\text{fission}} \rangle a \dot{\Phi}_R \quad (2.46)$$

where m is the fission deposit mass per unit area, " a " is the area of the deposit, and $\langle \sigma_{\text{FISSION}} \rangle$ is the fission cross section averaged over the incident neutron spectrum taken to be thermal in this case.

In addition to correcting for pulses due to alpha particles naturally occurring in the fission deposits, there are a number of corrections such as self-absorption in the fission foil, scatter from the foil backing, counting efficiency, etc. which must be made (Heaton, 1975; Gilliam, 1975).

For all of these cases the correction factor for a test neutron instrument placed at the calibration point in the reference field will be

$$CF = \frac{\dot{\Phi}_R}{(R_{\text{obs}}/t)_c N'(c)} \quad (2.47)$$

which for pulse counting field instruments is

$$N'(c) = (N_{\text{DT}} N_{\text{AS}} N_{\text{RR}} N_{\text{BKG}} N_{\text{T}})_c \quad (2.48)$$

If one is interested in dose equivalent rate instead of fluence rate one must use an appropriate conversion factor.

$$\dot{H} = C_{\Phi} \dot{\Phi} \quad (2.49)$$

For thermal neutrons

$$C_{\Phi} = 1.031 \times 10^{-9} \text{ rem} \cdot (\text{n} \cdot \text{cm}^{-2})^{-1} \quad (\text{NCRP 39, 10CRF20})$$

$$C_{\Phi} = 1.068 \times 10^{-9} \text{ rem} \cdot (\text{n} \cdot \text{cm}^{-2})^{-1} \quad (\text{ICRP 21})$$

Often the detectors described in this section are covered with cadmium. Cadmium will capture the thermal neutrons due to its large

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thermal cross section, so any observed response will be due to higher energy neutrons or to background, and other radiation effects.

Moderate Energy Neutrons

4) Fission chambers: This is the same as for the thermal case except the $\langle \sigma_{\text{FISSION}} \rangle$ is over the actual neutron spectrum in the moderate energy reference beam.

5) Remmeter: These are instruments which detect neutrons that are moderated to thermal energies. Their response is tailored in an attempt to match the curve of dose equivalent dependence on energy. Their output for a given spectrum can be calibrated in terms of fluence and the expression for the reference field using such a calibrated Remmeter will be the same as Equation 2.39 and 2.47 for a BF_3 counter.

Fast Neutrons

6) Long counter: For sufficiently intense beams fission chambers could be used as reference instruments. Remmeters could also be used. For accelerator produced neutrons, the neutron fluence can be monitored with a (dePangher) long counter (dePangher, 1966).

The observed count rate, C , of the long counter, corrected for dead time, is

$$\dot{C} = \dot{C}_{\text{scatter}} + \frac{\epsilon(\dot{S}_0/4\pi)}{(r + d_0)^2} \quad (2.50)$$

where $\dot{C}_{\text{SCAT}}$ is the count rate due to scattered neutrons and can be measured with a shadow cone, ϵ is the product of the detector efficiency and the area of the long counter, $\dot{S}_0/4\pi$ is the neutron emission rate into 4π , and d_0 is the distance to the effective center of the long counter. Based on data in BCS 0813, d_0 can be represented by

$$d_0 = (-0.26 + 0.12E + 0.05E^2) \quad (2.51)$$

with d_0 in cm and E in MeV.

When used with an accelerator, the long counter can be calibrated with a Cf-252 source of known emission rate (expressed as a known fluence rate at the surface of the detector). The count rate in the long counter due to the Cf-252 source will be

$$\dot{C}(\text{Cf}) = \dot{C}_s(\text{Cf}) + \frac{\epsilon_{\text{Cf}}(\dot{S}_0/4\pi)}{(r + d_0)_{\text{Cf}}} \text{Cf}$$

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The count rate due to the accelerator will be

$$\dot{C}(\text{acc}) = \dot{C}_s(\text{acc}) + \left(\frac{\epsilon_{\text{acc}}}{\epsilon_{\text{Cf}}} \right) \epsilon_{\text{Cf}} \frac{(\dot{S}_0/4\pi)_{\text{acc}}}{(r + d_0)_{\text{acc}}}$$

so the neutron emission rate in the accelerator beam is

$$(\dot{S}_0)_{\text{acc}} = \frac{(\dot{C} - \dot{C}_s)_{\text{acc}} (r + d_0)_{\text{acc}}^2 (\dot{S}_0)_{\text{Cf}}}{(\dot{C} - \dot{C}_s)_{\text{Cf}} (r + d_0)_{\text{Cf}}^2 \epsilon_{\text{ref}}} \quad (2.52)$$

where the efficiency of the long counter for the accelerator produced neutrons relative to Cf-252 is given in Table 2.13 (from BCS 0813). Note in Equation 2.52 the count rate due to scattered neutrons is not necessarily the same for the accelerator produced neutrons, and $\dot{C}_s$ should be measured in both cases with a shadow shield.

The correction factor for instruments placed in this beam will be

$$\text{CF} = \frac{\dot{\Phi}_{\text{acc}}}{(R_{\text{obs}}/t)_c N'(c)} \quad (2.53)$$

where $N'(c)$ is given by Equation 2.48 and the fluence rate,

$$\dot{\Phi}_{\text{acc}} = (\dot{S}_0)_{\text{acc}} [4\pi(r + d_0)^2]^{-1} \quad (2.54)$$

Instruments such as proton telescopes, associated particle counters, etc., for determining neutron fluence rate are beyond the scope of this chapter.

Known Field Technique

Thermal neutron sources

There are no naturally-occurring sources of thermal neutrons but they can be obtained by placing a moderator around a fast neutron source. However, for practical purposes the fluence rate is usually very small in this case. When making a thermal source in this matter it is very important to make measurements with and without a Cd cover to determine how well the source has been thermalized.

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TABLE 2.13 DE PANGHER LONG COUNTER EFFICIENCY RELATIVE TO THAT OBTAINED USING A ^{252}Cf SOURCE (1)

ENERGY MeV	RELATIVE EFFICIENCY	ENERGY MeV	RELATIVE EFFICIENCY	ENERGY MeV	RELATIVE EFFICIENCY
0.02	0.933	2.10	1.031	4.96	0.958
0.10	0.933	2.20	1.033	5.00	0.959
0.20	0.933	2.40	1.029	5.25	0.958
0.30	0.933	2.60	1.021	5.30	0.957
0.40	0.934	2.80	1.000	5.36	0.913
0.50	0.939	2.90	0.975	5.40	0.942
0.60	0.948	2.95	0.962	5.45	0.946
0.70	0.961	3.00	1.019	5.60	0.947
0.80	0.979	3.10	1.010	5.80	0.943
0.90	1.001	3.20	0.988	6.00	0.935
1.00	1.024	3.40	0.961	6.10	0.929
1.10	1.041	3.60	0.955	6.20	0.919
1.20	1.047	3.80	0.955	6.22	0.864
1.30	1.050	4.00	0.959	6.26	0.864
1.40	1.050	4.20	0.955	6.30	0.888
1.50	1.049	4.25	0.950	6.35	0.917
1.60	1.046	4.30	0.941	6.40	0.922
1.70	1.043	4.35	0.945	6.50	0.925
1.80	1.040	4.40	0.949	6.60	0.940
1.90	1.039	4.60	0.959	6.70	0.931
2.00	1.037	4.80	0.961	6.80	0.930
2.05	1.035	4.90	0.959	6.90	0.929
2.08	0.959	4.93	0.942	7.00	0.928

(1) From BSC 0813.

Moderated neutron sources

One way to achieve a neutron spectrum with a lower average energy than that of a fast neutron source is to place moderators of various material around a fast neutron source. Livermore Laboratory reported (Griffith et al., 1978) results with spheres of polyethylene, D_2O and Al surrounding the pneumatic source-transfer head for neutrons from Cf and PuBe sources. The fluence and dose rate for a 15 cm radius D_2O sphere surrounded by 0.02 inches of Cd (see Schwartz, 1980 for construction details) with a Cf-252 source at the center has been reported by Ing and Cross (1983). The reference field fluence rate for Cf with source strength $\dot{S}_0$ neutrons per second is

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$$\dot{\Phi}_R = 0.89 \frac{(\dot{S}_0/4\pi)}{r^2} N \quad (2.55)$$

where $N = N_A N_{AS} N_{SA} N_{RR} N_{BKG} N_{TN} N_{ND}$ and the factor 0.89 accounts for neutrons moderated to energies below the Cd cutoff. The attenuation coefficient in N should be averaged over the moderated spectrum.

The conversion factor for this spectrum is 9.0×10^{-6} mrem cm^2 , so the dose equivalent rate is

$$\dot{H}_R = 2.29 \times 10^{-3} \frac{\dot{S}_0}{r^2} N \quad (2.56)$$

where $\dot{H}_R$ is in mrem/hr, and $\dot{S}_0$ is in neutrons/sec.

Fast neutron sources

For a given source strength, the fluence in the reference field is

$$\dot{\Phi}_R = \frac{(\dot{S}_0/4\pi)}{r^2} N \quad (2.57)$$

where N is the same as for Equation 2.55. Hence the attenuation coefficient should be averaged over the Cf-252 spectrum. For Cf-252, the conversion factor from fluence to dose equivalent (Schwartz, 1982) is 3.33×10^{-5} mrem cm^2 , so

$$\dot{H}_R = 9.54 \times 10^{-3} \frac{\dot{S}_0}{r^2} N \quad (2.58)$$

with $\dot{H}_R$ in mrem/hr and $\dot{S}_0$ in neutrons/sec.

Effective Center Correction

There is one other important correction for calibrating neutron instruments with sources which has not yet been discussed. This is a correction for the effective center of the instrument being calibrated. For spherical neutron detectors and point sources, the fluence at the geometric center of the detector should be increased by a factor of approximately $(1 + (r/d)^2/6)$ where r is the radius of the sphere and d is the distance from the source to the geometric center of the detector (see Axton, 1972; Harrison, 1981). The correction factor for other geometry detectors or sources will be different.

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Beta Instrument Calibration

At present, the usual reference instrument for beta particle measurements is an extrapolation chamber. Due to the complex nature of this instrument, it is mainly used in a laboratory setting. Use of an extrapolation chamber to measure the output from beta sources allows them to be standardized in terms of absorbed dose to air or to tissue by applying appropriate stopping power ratios. To obtain uniform beams it may be necessary to use beam flatteners which in turn will affect the beta spectrum. Air absorption and scatter also affect the beta spectrum.

Reference Instrument Technique

The extrapolation chamber is an ionization chamber which behaves as a Bragg-Gray cavity in which the separation between the anode and cathode can be varied so measurements of ion current at various distances can be made and the results extrapolated to zero plate separation.

The absorbed dose rate to air, $\dot{D}_A$, is

$$\dot{D}_A = \frac{W}{e} \frac{k'}{A\rho} \frac{d}{dx} (i_u k(x)) \quad (2.59)$$

where W/e is the average energy to produce an ion pair in air per electric charge ($33.85 \text{ J/}^\circ\text{C} = 33.85 \text{ eV/ion pair}$; ICRU 31, 1979), A is the effective area of the collecting electrode, ρ is the density of air, x is the chamber separation (see Loevinger and Trott, 1966, for effects of chamber separation correction due to dependence on plate voltage), i_u is the ionization current due to ion rate for a chamber built with walls fully matched to the chamber gas with respect to atomic number, and k' , and k are correction factors (see Eur 7365, Böhm 1980, 1976).

Known Field Technique

A draft international standard recommends using C-14, Pm-147, Tl-204, Sr-90 + Y-90 and Ru-100 + Rh-106 sources as standards. It gives criteria for residual maximum energy based on range, on beam uniformity, and approximate dose rates at specified distance per unit activity with certain beam filters being present.

Table 2.14 lists recommended point sources (Owens, 1972), flattening filters, and the approximate dose rate at the calibration point per unit activity of the source.

Another common source to calibrate beta instruments is a uranium slab. The surface dose due to beta particles from a uranium slab is approximately 230 mrad/hr.

Additional information on calibration instruments for measuring short-range radiation can be found in IAEA 150 (1973).

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TABLE 2.14 BETA SOURCE CALIBRATION INFORMATION ⁽¹⁾

RADIONUCLIDE	CALIBRATION DISTANCE (cm)	SOURCE TO FILTER DISTANCE (cm)	FILTER MATERIAL AND DIMENSIONS	APPROXIMATE DOSE TO TISSUE PER UNIT ACTIVITY mrad·h ⁻¹ ·mCi ⁻¹
¹⁴⁷ Pm	20	10	1 Disc of polyethylene terephthalate of radius 5 cm and mass per area 14 mg cm ⁻² with hole of radius 0.975 cm at center.	23
²⁰⁴ Tl	30	10	2 Concentric discs, 1 disc of polyethylene terephthalate of 4 cm radius and mass per area 7 mg cm ⁻² plus 1 disc of polyethylene terephthalate of 2.75 cm radius and mass per area 25 mg cm ⁻² .	250
⁹⁰ Sr + ⁹⁰ Y	30	10	3 Concentric discs of polyethylene terephthalate each with mass per area of 25 mg cm ⁻² and of radii cm, 3 cm and 5 cm.	240

⁽¹⁾Based on Owens, 1972.

Alpha Instrument Calibration

Instruments for measuring alpha particles usually fall into one of two classes mentioned in Section 2.4.5, namely, field instruments used to monitor for the presence of alpha radiation, or laboratory instruments for measuring the alpha activity in specially prepared samples. The latter is covered very well in NCRP Report 58 and is beyond the scope of this chapter.

Instruments used in the field are normally calibrated using the known field technique.

Due to the short range of alpha particles in materials, there is significant absorption in the instrument walls, in air, and in the source itself. This means that even instruments calibrated in a specific geometry with a calibrated source may require large corrections when used in the field.

Sources used to calibrate alpha instruments should be standardized in terms of activity, activity per unit area, emission rate, or emission rate per unit area. Ideally, the source area should be larger than the detector area (see Ballard, 1981) so that the entire active volume of the detector is irradiated. If smaller sources are used it will be necessary to form some kind of averaging procedure to irradiate the entire active volume.

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2.5 CALIBRATION LABORATORY

The building space, facilities, equipment, staff and methodology necessary to properly operate a calibration laboratory depend on the volume and type of work undertaken. Calibration facilities exist that employ from a fraction of an employee's time up to tens of people. A brief description (taken largely from IAEA 133, 1971) of a typical medium-sized calibration laboratory is given as a guide to people that might have the responsibility of setting one up.

2.5.1 Building Space

The building for the calibration laboratory and the land on which it is located need to be closely integrated. There is a need to have a low scatter building so that calibration work may be completed with the required accuracy and that there be a low radiation level at external building walls. These conditions can be met by a relatively large building so that the inverse square law and moderate shielding in the outer walls will reduce the dose-rate from exposed radiation sources and radiation generating machines to acceptable levels at the exterior of the building walls. The construction of a moderate-size building on a large lot, which can have rigidly controlled access so that the external environment at the building wall need not be reduced to non-control levels, is preferred on a cost basis. In areas of high land cost, however, this alternative may not be preferred and a moderately large building with adequate external wall shielding may be required. In any case, the variation in the scatter at the positions of instrument calibrations should not exceed the variations due to instrument positioning on the test irradiation assemblies.

For certain types of work shielding is very necessary for reasons other than protection. Instruments that measure levels of radiation at or slightly above background are best calibrated in an area with low background. When other external radiation sources are nearby, additional shielding will serve to keep the background levels within the calibration cell constant.

Special features of the building should include shielded film and other personnel dosimeter detector storage areas of constant low background radiation and controlled temperature and humidity so that dosimeters can be stored before and after exposure. In the design of the laboratory building, flexibility in the use of all of its facilities should be provided. Maximum requirements include 1) irradiation rooms; 2) radiation source storage vault; 3) personnel dosimeter detector storage vault; 4) administration offices; and 5) receiving and shipping areas.

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2.5.2 Facilities

To carry out the calibration program, a laboratory requires properly designed and instrumented assemblies (sometimes called jigs or rigs) to position the test dosimeters, or portable instruments in proper relationship to the source. The number, size, and complexity of the laboratory assemblies will depend on the type of radiations, instruments, and dosimeters needed, and on the amount of calibration carried out. For laboratories that calibrate large numbers of the same type of beta-gamma survey instruments, calibration wells have the advantages of low exposure to the operators, of utilizing less floor space and of very little handling of the source.

Working conditions in the calibration laboratory should not cause excessive radiation exposure to personnel. Personnel exposure should be kept as low as practical and should in no case under normal operating conditions exceed levels allowed by regulation. To meet this condition, personnel shielding, remote instrument reading and positioning facilities, automatic source handling mechanisms, and other mechanical or remote operations are recommended.

2.5.3 Staff Qualifications and Training

The size and nature of the laboratory staff depend on the position of the laboratory in the organizational hierarchy (i.e., on the calibration accuracy required) and on the volume of the business. In a typical medium-sized laboratory the staff might consist of a director, a physicist, several technicians and a secretary. This provides a staff of two professional qualified persons to look after the scientific aspects of the laboratory work and several technicians to assist in the performance of the routine aspects of the laboratory calibration program. The technical integrity of the laboratory should be above reproach, and a competent staff of recognized capability is required to ensure this status.

The qualifications of each member of the laboratory should be carefully reviewed to ensure an experienced and capable staff. This type of laboratory should not be staffed with inexperienced personnel.

Ideally, the manager should have an academic degree in one of the sciences, mathematics or engineering, and should have a minimum of five years practical experience in personnel dosimetry and radiation monitoring programs. The manager should understand the basic principles involved and be competent in the use of the reference and other standards of the laboratory and its equipment. For example, competent use of reference ionization chambers, extrapolation chambers, R-meters, precision long counters and similar radiation measuring equipment should be understood so that proper calibration can be provided. The manager should possess a good understanding of statistics and the application of statistics and mathematical analysis to the quality control programs. Finally,

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the manager should have demonstrated good administrative capabilities, ability to issue clear and concise scientific reports, and work well with people.

The physicist should have earned a degree or an equivalent qualification in one of the physical sciences, mathematics, or engineering, and a minimum of two years experience with personnel dosimetry and radiation monitoring. In general, the physicist should have knowledge, experience and capability in all areas prescribed for the manager.

The technicians are required to assist in making the exposures. The capabilities and experience of the technician should lie in the field of physical sciences. Experience in calibrations and radiation protection or health physics activities is desirable.

A secretary who should have the qualifications established for advanced secretarial positions is required for the organization. Duties will include the preparation, filing and maintenance of the record copies of the calibrations as well as the conventional secretarial duties as directed by the laboratory manager and the physicist.

2.6 PRACTICAL DETAILS IN CALIBRATING INSTRUMENTS

All calibration procedures begin with the arrival of the equipment to be calibrated, continue with preliminary system checkout, calibration, and then end with certification and shipping of the equipment. This total procedure must be documented in a checklist form since every step is an important part of the calibration process. In the following sections, items that should be covered by such a checklist are reviewed.

2.6.1 Initial Set-up

Any radiation measuring equipment may be contaminated when it arrives at a calibration facility. Therefore, each item must be checked before being admitted to the calibration procedure. Generally this is done using a pancake proportional counter (Chapter 4) or GM counter (Chapter 5). To determine the presence of removable contamination, a wipe sample could be taken and counts from the sample detected by a gas flow proportional counter, solid state detector, thin window GM tube, etc. For an accurate determination of the amount of removable contamination present it will be necessary to identify the radionuclides on the wipe sample and correct the observed count rate for experimental conditions.

A more serious problem from the standpoint of health safety may be a gross contamination of an instrument by tritium, which normally is measured by a wipe sample counted in a liquid scintillation system. If an instrument came from a facility where large volumes of

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tritium are used or processed, such as a medical research facility, special procedures should be used to insure that the equipment arrived "clean." A similar situation arises when a stationary system is to be calibrated. A check of background should be done in order to be certain that no gross contamination is present that may 1) alter the accuracy of the calibration; or 2) contaminate the portable tools used in calibration (radionuclide sources, ionization chambers, screwdrivers, etc.). Also, a check for extraneous sources that may inadvertently be placed in a location that will affect the calibration is necessary.

All unshielded electronics are affected by radiofrequency interference, therefore it is prudent to check for the presence of large sources of radiofrequency radiation. For example, one should neither use nor calibrate a low level proportional counter in a room that adjoins an arc welding facility. Radio transmitters (including radar) are well known to cause interference to ionization chamber instruments.

Other obvious items should be checked in incoming equipment shipments, for instance:

1. Does the equipment belong at this calibration facility?
2. Are all parts present and are they matched?
3. Is the equipment clean (exterior appearance)?
4. Is anything broken (broken meter glass may signify rough handling and the chance of an intermittent failure that could affect calibration)?

Before initiation of the calibration procedure, appropriate paperwork should be assigned to each piece of equipment since the calibration is virtually useless if it is not documented.

It should be obvious that non-facility related equipment should be restricted. For example, an incident occurred in a nuclear research facility in 1965 where a researcher was disassembling an old sextant in a room adjoining the counting lab. His intention was to clean up this antique and display it in his home. He found some crusty white material near the reticule and removed it letting some of it scatter on the floor. After all, it was just corrosion! The next morning the counting lab was unusable because of external high background levels. "Corrosion" in the sextant was a radium compound used to illuminate the reticule at night. This radionuclide and its daughters were transported throughout the building not only by the shoes of the laboratory's personnel but also by the arrival and subsequent decay of the radium daughter "radon," a gas. Much time, money, and effort had to be committed in cleaning up the building. The counting lab from that time on had a new elevated quasi-permanent background level.

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2.6.2 Equipment Checkout

The proper procedure would include a complete check of both the electronics and mechanical functions of an instrument before beginning the actual calibration procedure. The batteries should be checked with an external battery checker and not merely with the "Battery Test" switch on the instrument being calibrated. This is also applicable to stationary instruments. For most instruments, detailed steps for accomplishing these tasks are incorporated into the operations manual by the manufacturer. They should be itemized on a checklist as part of the normal calibration procedure.

2.6.3 The Actual Calibration

Calibration involves a combination of electronics and ionizing radiation procedures. For example, the sensitivity of an ionization chamber to ionizing radiation is a function of 1) the voltage placed on the electrodes of the chamber (Chapter 3) in that it must be on the plateau region (which is a function of chamber gas content and electrode design and spacing) and 2) the proper adjustment of the various circuits that will measure signal current. But it is also a function of several characteristics that have no direct relationship to electronics but only to the incoming radiation, such as 1) the area of the chamber electrodes, 2) the construction of the chamber wall, and 3) the energy of the incoming radiation. Thus, both electronic adjustments and radiation exposure are required to facilitate the calibration of an ionization chamber. This principle applies to all radiation measuring equipment from a small TLD (thermoluminescent dosimeter) chip to the totally computerized GeLi semiconductor detector spectroscopy system.

Each instrument calibration at the laboratory should have a protocol stating explicitly what steps should be followed in its calibration. This protocol should include at least the following 1) how the standard is used (i.e., what techniques in Section 2.4.5 are used), 2) what in-house quality control procedures should be used prior to calibration, 3) what radiation sources are used and at what field strength the instrument should be calibrated, 4) what precalibration checks are needed, 5) what setup procedure should be used during the calibration, and 6) what corrections must be applied.

The raw data should be taken and recorded in such a manner that it can readily be retrieved and examined if in the future there is some question about the calibration of a particular instrument.

Since each step in the calibration procedure will contribute some uncertainty to the overall calibration, care should be taken at each step to ensure that the resulting total uncertainty does not exceed that claimed by the laboratory for calibrating that particular type of instrument.

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2.6.4 Possible Sources of Inaccuracy in Calibrating Instruments

Table 2.15 gives typical sources of inaccuracy encountered in instrument calibration. Most of these sources of inaccuracy were discussed in Section 2.4.5. In the ideal case all of the typical sources of inaccuracy listed, plus any others which are specific to the particular calibration set-up involved, will have been evaluated to arrive at an overall estimate of the accuracy of the calibration.

The possible sources of inaccuracy listed in Table 2.15 are primarily associated with "hardware." The table is subdivided into various areas associated with the calibration process (i.e., the survey instrument to be calibrated, the set-up apparatus, and the reference standard - either a radioactive source or an instrument). In addition to these errors, there are also errors associated with the "metrologist" such as:

- general laboratory procedures (carelessness, leaving unattended check sources about, etc.)
- mishandling or mistreating the apparatus
- failure to check that equipment is operating properly
- applying a calibration method not suitable for the instrument being calibrated
- misreading instruments and ancillary apparatus
- computational errors
- using incorrect values for correction factors

2.6.5 Calibration Report

The final product of the calibration is a calibrated instrument and its calibration report. This report should clearly state the calibration conditions. After the report is completed, it should be examined for obvious "blunders," i.e., misplaced decimal points, unreasonable correction factor for that instrument, etc. Ideally, the report should be examined by at least two staff members at the calibration laboratory before it is issued.

2.7 QUALITY CONTROL

Quality control ultimately is as good or as bad as the person in charge of the calibration facility wants it to be. The product of the facility is calibration and maintenance of radiation measuring equipment and nothing else. For this reason the equipment that leaves the facility should be in perfect operating condition, adequately documented and, of course, calibrated to a known accuracy.

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TABLE 2.15 POSSIBLE SOURCES OF INACCURACY IN INSTRUMENT CALIBRATION

1. TYPICAL SOURCES OF INACCURACY ASSOCIATED PRIMARILY WITH THE SURVEY INSTRUMENT.

Voltage supply instability
Insufficient electric field strength to collect all the ion pairs
Pulsed radiation field measurement
Stabilization time
Switching transients
Capacitance effects
Overload characteristics
Dead-time effects
Drift in instrument electronics
Radiation-induced leakage
Response in extracamerall volumes
Geotropism
Directional or orientational dependence
Response to stray RF or magnetic fields
Response to radiation scattered by the calibration set-up
Response to radiation scattered by the room
Response to other radiation (e.g., response to gammas emitted by a neutron source)
Attenuation in components of survey instrument
Vented ion chambers becoming sealed
Temperature and pressure corrections for vented ion chambers
Humidity
Energy response of instrument
Insufficient wall thickness for charged-particle equilibrium
Inherent imprecision of instrument
Statistical variation in amount of charge collect due to
 – fluctuations in number of incident photons, neutrons, etc.
 – variation in the energy of the primary charged particle produced by the incident radiation
 – Poisson distribution in the number of ion pairs produced
 – Instantaneous fluctuations of exponential discharge—charge characteristics of input circuit

2. TYPICAL SOURCES OF INACCURACY ASSOCIATED PRIMARILY WITH THE CALIBRATION SET-UP.

Reproducibility of placement of survey instrument in set-up
Length of time of exposure
Uniformity of beam across survey instrument detector
Uniformity of beam throughout active volume of detector
Using a collimated beam smaller than the detector
Air attenuation of the calibration beam
Uniformity of additional filtration
Kilovoltage of x-ray unit (both its value and stability)
Half-value layer measurement
Failure of field to follow inverse square law
Scatter from set-up apparatus
Energy spectrum change due to scattering within a calibration well

3. TYPICAL SOURCES OF INACCURACY ASSOCIATED WITH USING A CALIBRATED RADIOACTIVE SOURCE TO CALIBRATE THE SURVEY INSTRUMENT.

Uncertainty of reference source calibration
Radionuclidic impurity
Decay corrections
Attenuation in air
Attenuation within source and its holders
Anisotropic emission from the source
Scatter from source components (i.e., shutters, collimators, etc.)
Source not monoenergetic (this may be intentional for some calibrations)
Multiple types of radiation present (i.e., gammas and betas from Co-60)
Uncertainties in decay scheme (needed to evaluate the exposure rate constant)
Radiation leakage from source container

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TABLE 2.15 (Continued)

4. TYPICAL SOURCES OF INACCURACY ASSOCIATED WITH USING A REFERENCE INSTRUMENT TO CALIBRATE THE SURVEY INSTRUMENT.

Uncertainty in reference instrument calibration
Energy response of the reference instrument (particularly if calibration at one energy and used at a different energy)
Electrometer leakage current
Leakage of charged capacitor in capacitive—feedback electrometers
Ionization chamber electrical leakage
Power supply fluctuations
Electrometer (stability and accuracy)
Many of the possible sources of inaccuracy listed under survey instrument (Item 1) apply equally well here

A good quality control system will include:

1. Documentation of the maintenance and calibration steps that an instrument must undergo from entry to exit and a check sheet that indicates that these steps were taken.
2. A detailed description of each step so that instrument adjustment or calibration will be performed properly.
3. Availability of all tools required to accomplish the defined tasks.
4. An inspection system that assures the proper calibration and maintenance of each instrument.
5. A maintenance and calibration certification affixed to the instrument.
6. A system of inventory control that not only keeps track of the instruments in the calibration laboratory, but also all those assigned to be calibrated at the facility. This should include an automatic recall system so that calibration cycles are properly adhered to (usually every 6 months to 1 year depending on the instrument and conditions of its use).
7. Control over repair parts so that they meet the specifications set forth by the manufacturer.
8. An in-house program to monitor the constancy of the equipment used for establishing the reference radiation field.

If the above controls along with others that may fit a particular facility are adhered to, a fully operational and calibrated instrument will be delivered.

2.7.1 Frequency of Calibration

Apart from any legal requirements, it is recommended that an instrument should be calibrated at least once every 12 months, or more frequently under severe conditions of use. A newly purchased instrument should also be calibrated before putting it into service. In addition to determining the correction factor, this initial test should check the scale linearity and range-change errors. If the

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instrument has suffered any damage or has been repaired, then adequate checks should be made to ensure that the original calibration is still valid; if not, then a recalibration should be undertaken.

The reference instrument (standard) used by the calibration facility should also be periodically compared to an instrument in the next higher level of the standards hierarchy. In some countries intervals of periodic checking are determined by special regulations. Where no such regulation exists the frequency of the calibration depends on the nature of the reference instrument as well as upon the procedure adopted to check its constancy. Where the reference instrument is in the form of a detector, e.g., an ionization chamber, it should be checked again approximately once every twelve to twenty four months. For standard sources a very much longer period is acceptable.

2.7.2 Constancy Checks

Between the regularly scheduled calibrations the user should carry out constancy checks. Where possible these should be made with the same type of radiation for which the instrument is used. A jig incorporating a gamma source may prove to be useful in checking a gamma-beta instrument. Some instruments have built-in check sources that can be used to provide constancy checks.

Constancy checks are doubly important when the instrument is used in high level radiation areas or when the instrument is used only occasionally. The leakage radiation from a reactor or accelerator operating under a fixed known condition can also be used to see if the instrument sensitivity has changed since it was last calibrated. If a change occurs the instrument should not be re-adjusted to give the original constancy check reading since external factors could have changed its characteristics. This is better done at the calibration laboratory where a wider range of standard sources may be used to check the sensitivity over the instrument's complete range.

2.7.3 Mechanical, Electrical and Environmental Evaluation

Performing a full "type-test" evaluation of an instrument will assist in the proper selection, operation and calibration of radiation monitoring instruments. The limitations of an instrument should be made known to the users, since they may often be as important, or more so, than its good features.

Calibration in the radiation environment alone does not constitute the complete "type-test." Other significant influences which should be included are (see also IAEA 133):

1. Mechanical effects:

- a) Shock and moisture resistance.

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- b) Resonance effects (vibration).
- c) Geotropism.
- 2. Normal environmental effects:
 - a) Temperature dependence.
 - b) Humidity dependence.
 - c) Atmospheric pressure.
 - d) Capacitance effects.
 - e) Chemical degradation.
- 3. Electromagnetic interference effects:
 - a) Magnetic fields.
 - b) RF fields.
 - c) Electrostatic fields.
 - d) AC line transients and noise.
- 4. Extra-cameral effects.
- 5. Electronics design:
 - a) Power supply stability.
 - b) Battery life.
 - c) Linearity.
 - d) Sensitivity.
 - e) Switching transients.
- 6. Human engineering.

The references in Section 2.1 give additional information on "type-testing" instruments.

The ideal instrument would be designed to operate in the presence of all of the above effects and still provide accurate readings. This, of course, is never possible but with some additional expense most of the normal problems can be alleviated. This is the reason why, for example, the majority of military survey meters are very different in both physical and electronics design from commercial instruments of equivalent apparent capability. Notably, military instruments are very resistive to mechanical and environmental influences.

2.8 TRACEABILITY AND QUALITY ASSURANCE

The concept of traceability is both important and elusive. Indeed, after many years of debate by recognized authorities there still does not exist a single definition of traceability that is acceptable by the entire scientific community (Belanger, 1980). A favored definition (op.cit.) is the following:

"Traceability to designated standards (national, international, or well characterized reference standards based upon

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fundamental constants or nature) is an attribute of some measurements. Measurements have traceability to the designated standards if and only if scientifically rigorous evidence is produced on a continuing basis to show that the measurement process is producing measurement results (data) for which the total measurement uncertainty relative to national or other designated standards is quantified."

It should be noted that traceability is sometimes also defined in a way that emphasizes NBS calibration of an artifact (instrument, source), or calibration against another standard in a chain or echelon of calibrations, ultimately leading to a calibration performed by NBS. A comparison of these two definitions provides perhaps the clearest exposition of two contrasting views of traceability, the one stressing characteristics of measuring instruments or standards, the other stressing requirements relating to quantifying measurement uncertainty. The one regards accuracy as a property of an instrument, whereas the other focuses on the quality of measurements.

A basic requirement of any measurement or calibration is consistency with the national standards. Measurement consistency can be demonstrated or implied. The user or regulating authority must ultimately decide whether implied consistency is adequate, or whether it must be demonstrated. In either case, some action or actions must be taken if consistency is to be achieved. If these actions are performed properly the measurement is considered to be consistent with the standard. It is essential that the steps taken be documented. Such documentation is the only evidence that the actions were taken and were appropriate.

2.8.1 Instrument Traceability

In those cases where implied consistency of a measurement with a standard is sufficient, it can be achieved through a calibration process (Eisenhower, 1982). The instrument used to make the measurement is calibrated by comparison with the appropriate national standard, either directly or indirectly through intermediate calibrations as shown in Figure 2.10. This is the traditional type of traceability. In some cases it is the only kind that can be realized because of inadequacies in the measurement support system. Radiation sources are frequently utilized in the instrument calibration process and should therefore also be calibrated in a manner that provides traceability to the appropriate national standard.

A shortcoming of instrument traceability lies in not being able to demonstrate that the measurement made with a traceable instrument is indeed consistent with the national standard. The consistency of the measurement must be implied inasmuch as the traceability chain ends with the instrument (Figure 2.10). Under favorable conditions a measurement made with a traceable instrument may be consistent with a

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standard. Some degree of uncertainty about the validity of the result remains, however, since the quality of the measurement itself has not been demonstrated.

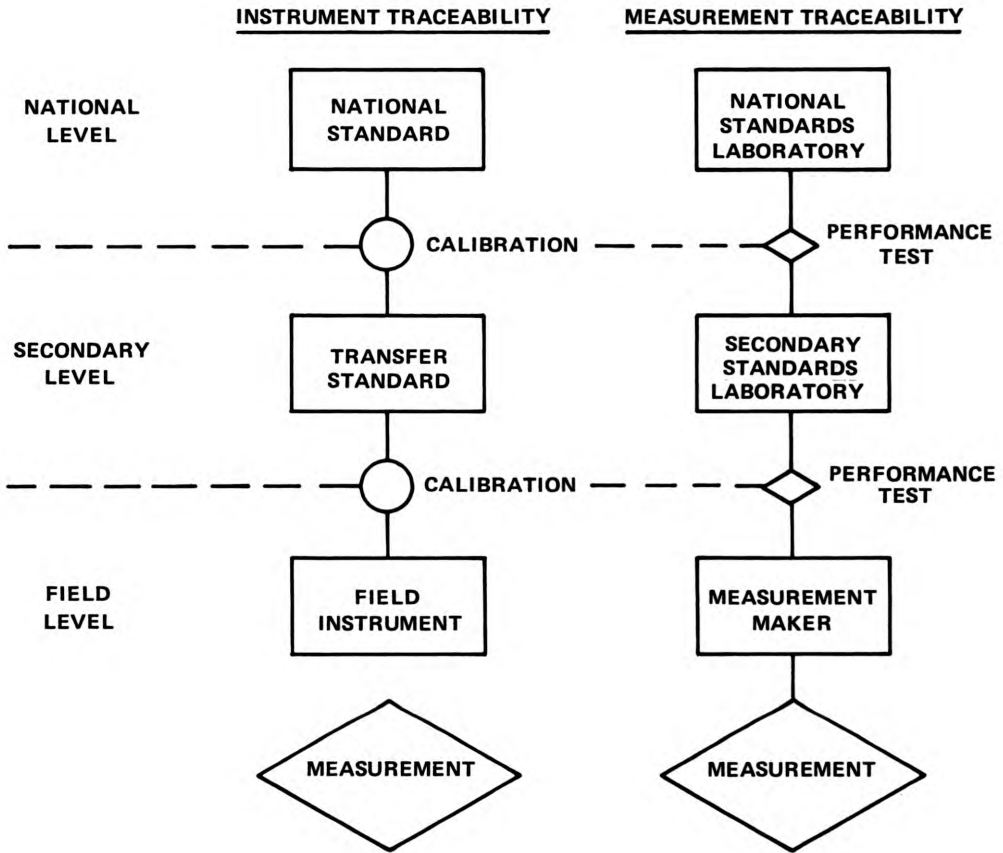


FIGURE 2.10 TRACEABILITY CHAINS FOR INSTRUMENT TRACEABILITY AND MEASUREMENT TRACEABILITY (EISENHOWER, 1982).

2.8.2 Measurement Traceability

For demonstrated consistency of a measurement with a standard the traceability chain must extend beyond the instrument to the measurement (Figure 2.10). In this case, traceability becomes a characteristic of the measurement itself, and there is documented evidence that the measurement is consistent with the appropriate standard. Measurement traceability is the most desirable type of traceability, since it is based on a demonstration that the complete measurement process is performing as intended, including the instrument, its user, and the procedures.

Measurement traceability is usually achieved by using a transport standard. This standard may be in the form of a radiation

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source, instrument or a dosimetry device that originates from NBS for the particular measurement under consideration. When a radiation source or instrument is used, its output or response is measured by the participant who received it, and the result is reported to the originating laboratory. If the reported value compares favorably with the value determined by the originating laboratory, a statement of satisfactory performance is provided to the participant.

When a dosimetry device is used as a transport standard, it is sent to the participant who gives it a nominal dose (or exposure). The dosimeter is then returned to the originating laboratory, where the dose (or exposure) is evaluated. If the participant's nominal measured value is within the acceptable range of uncertainty, a statement of satisfactory performance is provided. An important advantage of measurement traceability is that it can be achieved without the need of a transfer standard (instrument or source) that has been calibrated by NBS.

2.8.3 Measurement Quality Assurance

It is important to stress at the outset that the resolution of the issue of whether or not a particular system for realizing traceability is effective depends entirely on the real intent of traceability, i.e., to insure measurement of adequate accuracy. As indicated above, it is evident that mere calibration of a standard instrument by NBS, although perhaps necessary, is not sufficient for the task at hand. What is required for traceability is to demonstrate that the measurement process indeed produced "measurement results (data) for which the total measurement uncertainty relative to national or other designated standards is quantified." This key step demands careful definition of the quality assurance requirements for the particular measurement under consideration, and implementation of a systematic procedure to affect the desired demonstration. For ionizing radiation, the systematic procedure is commonly referred to as a Measurement Quality Assurance (MQA) Program, i.e., a program that allows one to demonstrate that the total measurement uncertainty including both random and systematic components of error relative to national or other designated standards is quantified, and sufficiently small to meet the requirements of the measurement process.

The development of a good MQA program requires a clear definition of the objectives, i.e., what is to be measured and what accuracy is required. A second step is to develop a model of the physical measurement process, i.e., what are the factors that affect the precision and accuracy of the measurement results? Next, the development of a suitable transport standard is a critical part of the development of a MQA program. The transport standard is a device, artifact, or material that must be stable, rugged, and well-characterized, and whose value is accurately known relative to national standards. Between measurements made on the transport

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standard at NBS it can be sent to the laboratories participating in the MQA program so that they can make measurements on the transport standard and thereby determine biases relative to NBS. Essential requirements for a transport standard are transportability and predictability.

For some measurements it is necessary for NBS to design and build a transport standard. In other cases it is possible to find a commercial instrument or device that is sufficiently stable to serve as the transport standard.

Finally, development of proper methodology is also very important. NBS generally makes recommendations to new users of MQA services as to good operating practices and provides consulting help to the participants as needed, so that each participant can have the best possible measurement quality assurance procedures within the capabilities of the laboratory. Users of MQA services should also conduct constancy checks that will monitor the stability of their reference standard and measurement or calibration procedures, and provide data on precision.

Measurement quality assurance services have been developed in the field of ionizing radiation that are suitable for use with photon dosimetry, electron dosimetry, and radioactivity measurements.

If all of the MQA program procedures recommended by NBS are followed, not only will the laboratory know when a problem arises, but they will also have data available that will make it possible to locate the source of the problem readily and correct it. Thus, participation in a MQA program provides feedback to permit self-correction action by the participant.

It is not uncommon for a laboratory participating in a MQA service for the first time to uncover a systematic error in their measurement process that has gone undetected for years. The other situation also surfaces, in which a laboratory has made conservative estimates of their measurement uncertainty and then finds after participating in a MQA program, that they can document an uncertainty for their measurement process that is much better than they previously estimated. The important point is that a MQA program provides a way for a laboratory to rigorously quantify its measurement uncertainty relative to national standards, and by so doing capture the elusive traceability.

2.9 LEGAL AND ETHICAL IMPLICATIONS

Failure to follow the best available practice in a radiation instrument calibration laboratory compromises the radiation protection of the very personnel for whose protection the calibration laboratory exists. Less than best effort cannot be condoned on ethical grounds, and will most likely lead to unnecessary, time consuming and expensive law-suits. There is indeed no substitute for following well thought out procedures that are updated as new

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information, in the form of improved measuring instruments and sources, better calibration techniques and/or intervals, revised maximum permissible dose levels, etc., is received. It is important to realize that following technically good calibration procedures is not sufficient. These procedures must be accomplished by quality assurance programs that demonstrate that the desired accuracy is indeed being realized. Informed, conscientious operation of a calibration laboratory will satisfy existing legal requirements (Chapter 12), and will reflect a high degree of professional ethics.

2.10 SUMMARY

The basic principles necessary to an informed approach to the calibration of the most common ionizing radiation measuring instruments were presented. This was accomplished by first describing the organizational structure of the ionizing radiation measurement community and the hierarchy of radiation standards. Emphasis was placed on the national standards which are based on fundamental physical constants. Less detail was devoted to the various types of transfer standards obtained by comparison with the national standards.

The two principal technique (known field and reference instrument) for calibrating instruments were discussed. General features of radiation sources used for calibrating instruments for x rays, gamma rays, beta particles, alpha rays and neutrons were reviewed. This was followed by a summary of the various types of instruments used for measuring radiation and which calibration techniques are commonly used to calibrate various types of radiation measuring instruments. There was a lengthy discussion of how the reference field at the calibration point and at the time of calibration is determined by both calibration techniques and for various radiation types.

General features of a typical medium scale calibration laboratory were described. This was followed by practical details of instrument calibration, including an estimate of the inaccuracies in the calibration process.

Quality control was then discussed in the context of calibration intervals, constancy checks and mechanical, electrical and environmental evaluation. A brief treatment of safety precautions was given. The important concepts of artifact and measurement traceability were outlined and the need for a national measurement support system to facilitate traceability demonstration was indicated. Finally, the legal and ethical implications of good (or bad) calibration practice were presented.

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REFERENCES

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CHAPTER 3

IONIZATION CHAMBERS

W. H. ELLIS and R. A. WALTERS

3.1 INTRODUCTION

The ionization chamber is one of the oldest and most relied upon instruments for radiation measurement. With proper design, the spectral response of the ionization chamber can be made very close to that of tissue. The ionization chamber can be designed to measure a wide spectrum of photon radiation and particle types: gamma-rays, x-rays, alpha particles, beta particles, neutron radiation, as well as high energy particles from machine sources. Among the gas-filled detectors, the ionization chamber is unique because no gas amplification of the initial ionization takes place. Due to recombination, the number of electron-ion pairs collected by the electrodes is equal to, or less than, the number produced by the interaction of the ionizing radiation with the gas. Because of this lack of gas amplification only very small currents are produced which limits the low radiation level detection capability of the ionization chamber. They are, however, commonly used as secondary calibration standards as a result of their simplicity and ease of fabrication as high quality instruments. Many details which are omitted in the following discussion can be found in classic books and monographs on ionization chambers by Rossi and Staub (1949), Curran and Craggs (1949), and Wilkinson (1950). More current and specific information on chamber design characteristics can be found in monographs and texts by Franzen and Cochran (1962), Price (1964), and Knoll (1979).

3.2 PRINCIPLES OF OPERATION

In Chapter 1, the theory of electrical conduction in gases and gaseous discharges was discussed. The ionization chamber operates in the plateau region of the current-voltage characteristic curve (Figure 1.16). Within the plateau, or saturation region, ion-electron recombination and diffusion losses can be made small. Consequently, for each ion pair generated by the interaction of an ionizing radiation particle with the chamber gas, an electron-ion pair is collected by the electrodes. The resulting electrical current is proportional to both the gamma-ray energy absorbed in the

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wall of the ion chamber and the electrons penetrating the ion chamber walls and absorbed in the chamber gas. The number of ion pairs produced per unit energy deposition into the gas is independent of the incident photon energy. Therefore, ionization chamber response is directly proportional to exposure (Roentgens for x- and gamma rays) and for reasons pointed out above, can also be used to measure absorbed dose in tissue (Grays). This relationship is only true over the energy region where the ionization chamber is in electron equilibrium.

The most basic ionization chamber configuration is the guarded air-wall, flat plate design, shown in Figure 3.1 (Knoll, 1979). The sensitive volume is defined by the electric field between the collecting plates and the collimated radiation beam. In the "cavity" region, interaction of radiation with air takes place and produces a current which flows through a high megohm resistor, R. The voltage produced across R (Figure 3.1) is measured by a sensitive voltmeter (Section 3.3.1) having an input impedance greater than R. The guard plates (or rings, if the chamber is cylindrical) contribute in defining the sensitive volume of the chamber by compensating for electric field distortion, shielding against the accumulation of charges on the insulators, and reducing leakage currents through the anode and cathode insulators. In those designs where the collector plate is maintained near ground, a change in the potential of the guard ring with respect to the collecting electrode will cause a change in the active volume of the chamber. Therefore, any charge collection which affects the guard ring will potentially result in measurement errors (Boag, 1966).

A free-air ionization chamber with an active volume of 10 cm³ will generate a current of 9.27×10^{-13} amp when irradiated by a 1 Roentgen/hr field. The current can be expressed by the following equation:

$$I = 9.27 \text{ XV} \left[\frac{P}{760} \frac{273}{T} \right] 10^{-14} \quad (3.1)$$

where I = current (amps), X = exposure rate (R/hr), V = active volume (cm³), P = pressure (mm Hg), and T = temperature (K). Ionization chamber currents are generally quite small. Special techniques required for their measurement are described later in this chapter.

To maintain electron equilibrium, the thickness of the air wall surrounding the sensitive volume must be increased when the photon energy is increased because the dimension of the air wall must be greater than the range of secondary electrons. Otherwise, the gamma-ray radiation of ⁶⁰Co (1.25 MeV), would require an air wall of four meters radius around a cylindrical chamber. As this is impractical in most cases, the free-air ionization chamber is restricted to measurements of low energy x-ray and gamma-ray sources (and is indeed the standard method of measuring low energy x-ray and gamma-ray exposure).

In order to bring an ionization chamber into electron

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equilibrium in a size not more than a few centimeters in diameter, the chamber cavity can be surrounded by a solid material which has nearly the same radiation interaction properties as air. For example, Bakelite, Lucite, and other plastics which have a mean atomic number close to air, can be used. This type of chamber is called a cavity, or thimble, ionization chamber and is the most common design for general survey and calibration work. Typical examples of this type of chamber are described in the next section.

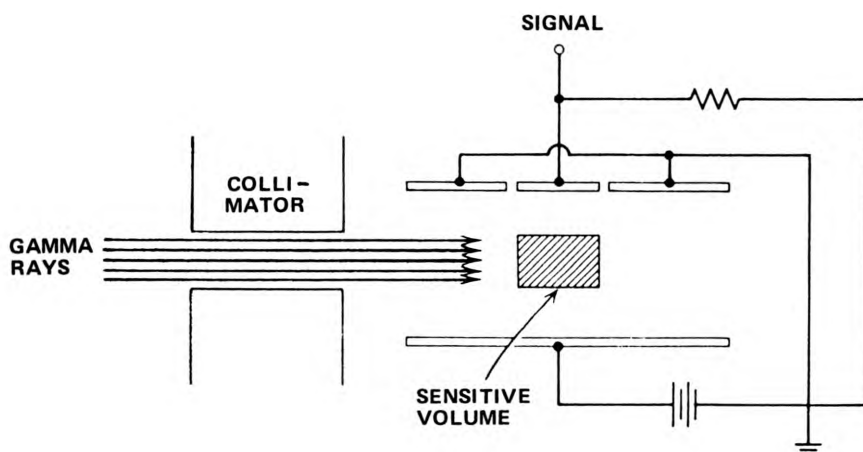


FIGURE 3.1 PARALLEL PLATE GUARDED AIR-WALL IONIZATION CHAMBER.

Many ionization chambers use air as a fill gas and are open to the atmosphere. Therefore, corrections to exposure measurements are required as a function of air temperature and pressure. Each of these environmental characteristics affects the density of air, and thus changes the magnitude of the interaction of the radiation with air. Studies at NBS have shown that corrections for humidity are extremely small and can be neglected in most measurements. Corrections for atmospheric effects (pressure and temperature) are included in the bracket term in Equation 3.1.

To measure low levels of radiation, such as environmental levels, ionization chamber designs sometimes employ sealed chambers filled with a variety of gases. Argon at pressures up to 30 atmospheres is common. Sealed chambers do not require compensation for temperature and pressure. They will exhibit an enhanced sensitivity (per unit volume) as compared to atmospheric air chambers due to the increased absorption provided by the dense high pressure gas and the gas ionization resulting from radiation absorbed by the chamber walls. Large volume versions of high pressure chambers provide the capability of measuring exposures as low as a few microroentgens, while the typical atmospheric air chamber is limited to exposure measurements of one milliroentgen.

Up to this point mean-level ionization chambers operating in

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the current mode have been described. The ionization chamber can also be used in the pulse mode as a counter or pulse detector for energy resolved measurements of alpha and beta particles and fission fragments. But pulse counting in an ionization chamber presents a problem. The pulse height produced by an ionization chamber (such as Figure 3.1) is a function of the distance between the point where the radiation interaction forms the ion pairs and the point on the anode which collects the electrons. More specifically, the time dependent voltage drop across the megaohm resistor R in Figure 3.1 is a function

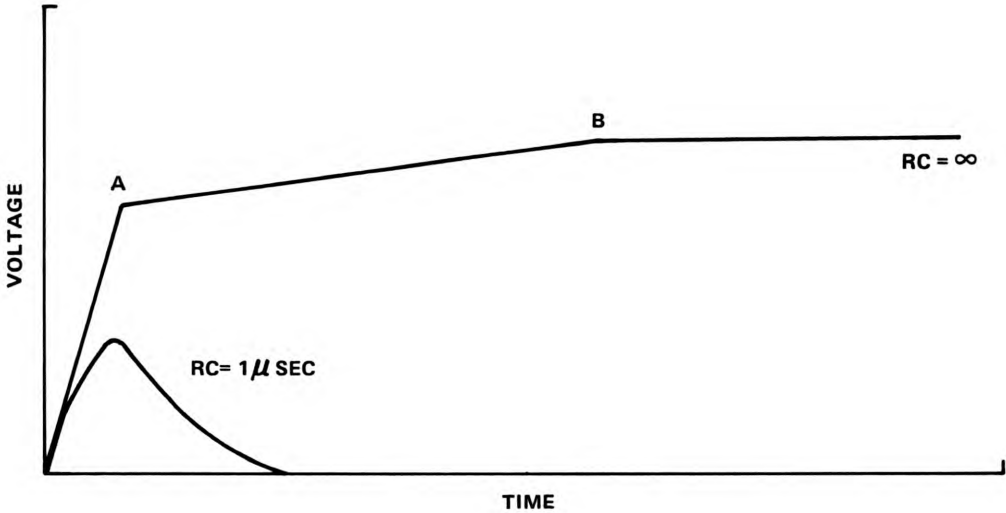


FIGURE 3.2 VOLTAGE PULSE ON COLLECTION ELECTRODE OF A PULSED IONIZATION CHAMBER.

of the system capacitance C and R . The lower curve in Figure 3.2 represents the response for an RC time constant of 1 microsecond in width. If the RC time constant could be made very long, one could view the arrival of each type of charge carrier. This case is represented by the upper curve in Figure 3.2. The fast rise of this curve to point "a" represents the collection of electrons by the anode and the subsequent slow rise to point "b" is due to collection of positive ions by the cathode. If an RC time constant is chosen equal to the electron collection time, usually 1 microsecond, the output pulse will also be nearly 1 microsecond in width. This pulse is reasonably short and allows a resolving time sufficiently short for pulse overlap not to be a major problem. However, using such a short time constant, the positive ions are not collected and induce a charge on the anode. This charge effectively reduces that due to electrons and thus reduces the pulse height. If the radiation induced ionization events are near the anode, there will be maximum

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effect, while near the cathode there is almost no effect. This undesirable effect may be alleviated by the addition of a grid that shields the anode from the positive ions as in the "Frisch grid chamber" (Price, 1964). Due to the extremely low pulse height of this type of chamber, however, sensitivity for low energy radiation is limited. As a result, pulse height measuring ionization chambers have, for the most part, been replaced by proportional counters. (A more favorable geometry for ion chambers, without an internal grid structure, is available with gas amplification in the proportional region which allows the use of much less sensitive electronics.)

3.3 IONIZATION CHAMBER INSTRUMENT DESIGN

Ionization chambers are fabricated in many different configurations. For descriptive purposes they can be grouped in three principal classes:

1. Survey instruments (mean level).
2. Laboratory instruments (both mean level and integrating) that are normally used in a fixed location.
3. Integrating ionization chambers.

Several common designs of these three classes will be described and their use for radiation measurement will be reviewed in this section. Ionization chambers designed for portable survey instruments should meet the specifications set forth in ANSI 13.4-1975 and additional specifications of the particular user, i.e., military. Pocket ionization chamber dosimeters are covered under ANSI N322-1977. The specifications cover what is to be included in the operation and maintenance manual, how and under what environment the instrument is to be tested, and how well the instrument can be expected to perform. Not all manufacturers follow ANSI standards, nor is it implied that they should. When these standards are followed, however, certain important design parameters will be met. The minimum requirements and the operational characteristics and methods for instrumentation of this type for radiation protection use are set forth in ICRU 20 (1971), NCRP 57 (1978), and for radioactivity measurements in NCRP 58 (1978).

3.3.1 Small Current Measuring Techniques

The sensitivity of mean level (rate measuring) ionization chambers (as opposed to an integrating type) is determined by the capability of electronic circuits to measure the chamber current. Typical currents in mean level ionization chambers vary from 10^{-6} amp in nuclear reactor control systems (neutron detection) down to 10^{-16} amp in measurement devices for low level radiation sources

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(radionuclides). Below 10^{-8} amp, standard circuitry does not provide a sufficiently high input impedance. For example, if an ionization chamber operating at 700 volts delivers 10^{-16} amp, the internal resistance of the detector is 7×10^{14} ohms, i.e., about the same resistance as an insulator. For such large values of impedance, standard electronic circuitry is not adequate and an instrument called an electrometer is required.

Three common electrometers for measuring small currents are:

1. Low grid current electrometer tube.
2. MOSFET solid state electrometer or associated solid state techniques.
3. Vibrating reed electrometer.

Each type of electrometer is connected to the chamber as shown in the block diagram in Figure 3.3. The grid of the electrometer tube is directly connected to the center electrode of the ionization chamber. The magnitude of the signal is adjusted by switching high megaohm resistors. The advantages of the electrometer tube are low cost and very low noise. The disadvantages are sensitivity to mechanical shocks (microphonics), poor stability (as much as 50% drift in the first hour requiring continuous "zeroing" when in use), poor temperature response, and short battery life (as with all vacuum tubes). MOSFET electrometers have essentially replaced electrometer tubes in most applications since they feature small size (can be located in the chamber), low cost, extremely low mechanical shock sensitivity, long battery life, and extremely high reliability. However, they suffer from high noise, high offset current, and poor overload characteristics (the MOSFET electrometer is easily damaged).

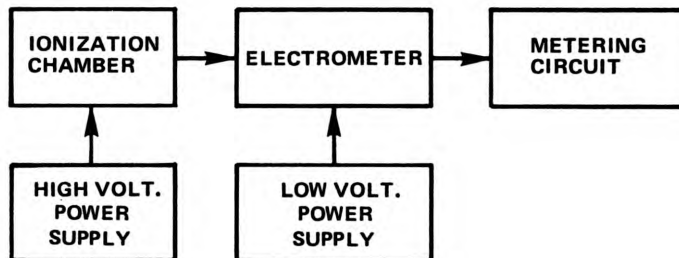


FIGURE 3.3 IONIZATION CHAMBER SURVEY METER DIAGRAM.

In some situations, the vibrating reed electrometer is the best choice. It has excellent stability, low noise, exceptional overload capabilities, almost no offset current (zeroing), and low temperature sensitivity. On the other hand, it is expensive, bulky and has complex circuitry giving rise to short battery life. In all

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of these systems, regulation of noise-free chamber voltage is not critical due to the existence of the broad ionization chamber saturation plateau. However, since the electrometer must have high gain, accurate regulation of the electrometer power supply is essential.

With the advent of large scale integrated circuits, inclusion of complex circuitry into a small volume is now possible. Electrometers are now using this technology. Complex solid state systems (MOSFET) are also being developed for inclusion into commercial and military ionization chamber instruments. Examples are varactor bridge circuits with amplification on a modulated signal and charge quantization feedback circuitry (Shapiro, 1971).

Negative feedback can be used to reduce the response time of high impedance electrometers. Otherwise they can exhibit very long response time. In some cases these response times can be as long as 45 seconds (to reach full scale) with 3 to 20 seconds being common.

Small current measurement systems and techniques are covered extensively in monographs such as Rossi and Staub (1949), Franzen and Cochran (1962), Faustein (1962), Price (1964), Knoll (1979), ICRU 20 (1971), and NCRP 57 (1978).

3.3.2 Survey Instruments

Survey meters are general purpose instruments used for ruggedized survey, as well as for specialized measurements. For example, a rf-shielded ionization chamber can be used to measure radiation from television sets, accelerators, radar sets, or other types of radio frequency generators. Typically these instruments use vibrating reed electrometer circuitry. They have response times on the order of 12 to 20 seconds (zero to full scale). Since in many of the instruments the ionization chamber is located internally or compactly adjacent to the case, range is an important factor to consider when measuring low energy sources. These instruments may also have thin windows on the bottom of the case (Juno configuration) for alpha particle and beta particle survey. Energy response characteristics for portable survey instruments are gamma > 7 keV, beta > 100 keV, and alpha > 4 MeV. The general characteristics and applications of survey ion chambers are covered by Price (1964), Boag (1966), Kiefer and Maushart (1972), Knoll (1979), ICRU 20 (1971), and ICRU 57 (1978).

Ionization chamber survey instruments have a linear energy response over a wide range (10 keV to 1.2 MeV, $\pm 10\%$) which includes most scattered photons. When properly designed they provide electron equilibrium operation up to about 1.2 MeV. Many portable ionization chambers are equipped with thin windows for measurement of beta particle and alpha particle sources. As mentioned previously, sealed survey instruments do not require temperature and pressure corrections.

In order to provide accurate directional dependance from area

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sources, a 2π geometry "panoramic" survey instrument was developed. This device uses an unsealed ion chamber consisting of a center electrode and a coated, expanded polystyrene outer electrode. Figure 3.4(a) shows the energy response from 6 keV to 1.25 MeV. Electron equilibrium is established with an air equivalent equilibrium cover, or cap, in place from 10 keV to 1.25 MeV. The equilibrium cap also provides some directionality that can be useful in locating radioactive sources.

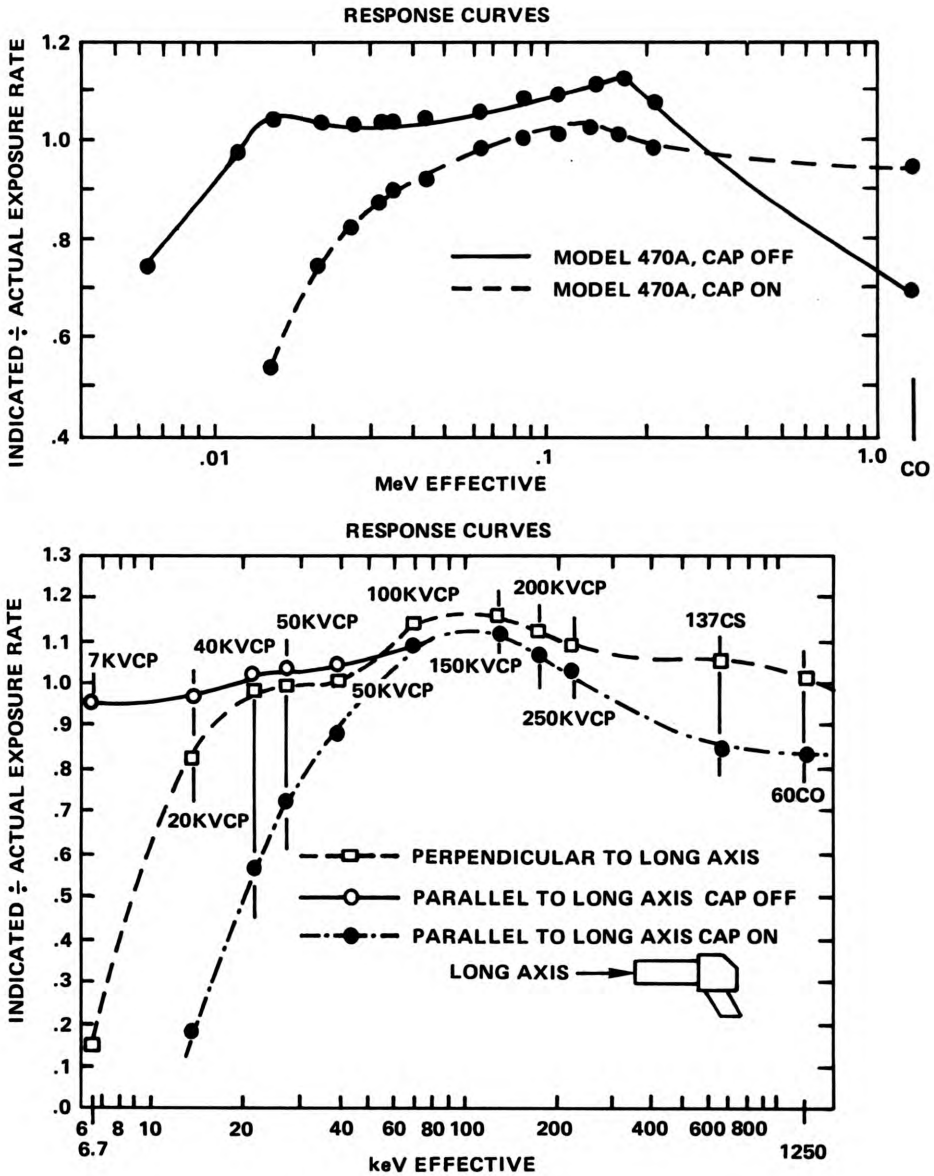


FIGURE 3.4 ENERGY RESPONSE FOR ION CHAMBER SURVEY METERS.

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Another common configuration employing a cylindrical chamber is called the "cutie pie" or, by some manufacturers, high Q, 2π . This inexpensive instrument has gained widespread application in industry. It has a thin window on the end to allow for beta particle and alpha particle measurements. Although it exhibits 2π geometry, it is most accurately calibrated from the side and should preferably be used in that geometry since the chamber is larger in diameter than it is long. As can be seen in Figure 3.4(b), a 17% drop in indicated exposure is experienced when the source axis is parallel (instead of perpendicular) to the chamber cathode. The "cutie pie" does not have any directionality. These are inexpensive instruments. They incorporate solid state electronics but require "zeroing" of the electrometer (generally MOSFET) at very short time intervals in order to maintain accuracy.

Modern instruments may be equipped with liquid crystal displays for operation in any position without meter misalignment (geotropism), and with auto ranging. Several new configurations are now being developed using miniature high pressure ionization chambers (Raymond and Nirschl, 1978). These chambers are able to measure exposure and have a uniform energy response down to less than 80 keV. All these devices are miniaturized by using integrated circuits. In some cases, the MOSFET electrometer and preamplification circuits are internal to the ion chamber. Dose rate readouts are usually digital with audible indication. Dose integration is also available. Because the sensitivity of these chambers is limited to greater than 10 mR/hr, plug-in or internal GM probes are sometimes provided to allow survey detection of low level sources. Further miniaturization using solid state detector "dosimeters" is under development with preliminary versions currently available.

3.3.3 Laboratory Instruments

Laboratory ionization chambers are usually designed for a single use, e.g., in-house reference instruments to be used in calibrating other instruments, the measurement of moderate level radiopharmaceuticals (the dose calibrator), the measurement of high level neutron fluxes in a nuclear reactor (^{10}B coated compensated ion chamber), etc. Ionization chamber instruments of the laboratory class are described in monographs such as Boag, 1966; Knoll, 1979; and NCRP 58, 1978

Paired ionization chambers (two identical chambers), one sealed and one open to the environment, are used to sample air for the presence of low energy beta particle emitting radionuclides such as tritium or ^{14}C . This system is especially useful in locations having a high radiation background such as nuclear reactor installations. Figure 3.5 is a block diagram of such a system. A measured flow of filtered air is supplied to one chamber in which both the general gamma-ray background and low energy emission from

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airborne particles or gaseous activity are measured. A second identical chamber is sealed and measures only the background activity. After input to a differential amplifier or current comparator, only the ionization current produced by the tritium, ^{14}C , or other low energy radionuclide source will be measured. Normal uncertainties are $\pm 10\%$ of the measured value, with backgrounds up to 5 mR/hr. A sensitivity of better than $1 \mu\text{Ci}/\text{m}^3$ for tritium can be obtained (Peetermans, 1972). The air is filtered before entry into the chamber because particulates, especially smoke, drastically decrease the ionization chamber current because of electron attachment (electron attachment is used as the basis of some ionization smoke detectors).

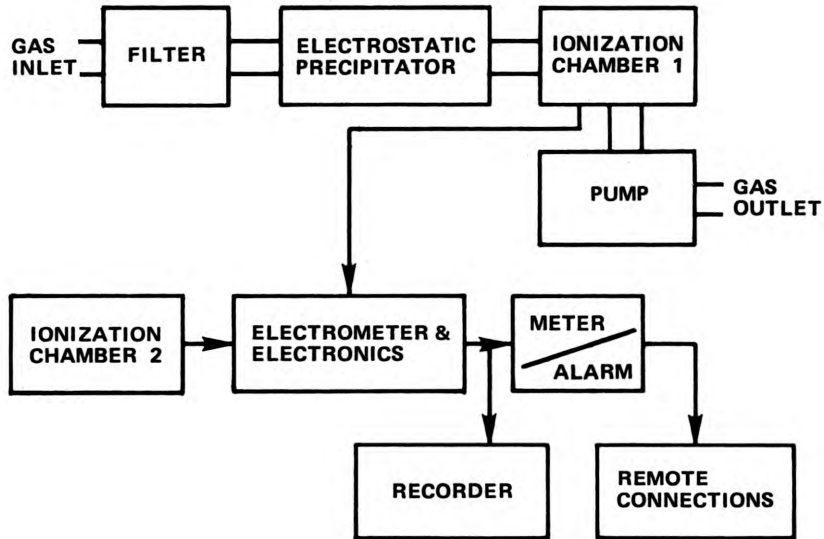


FIGURE 3.5 BLOCK DIAGRAM OF A LOW ENERGY (TRITIUM) AIRBORNE RADIONUCLIDE MONITOR.

The compensated neutron ionization chamber shown in Figure 3.6 can be employed to measure thermal neutron fluxes in very high gamma radiation environments. One set of plates is coated with ^{10}B and the interaction of neutrons with the coating produces 1.47 MeV alpha particles and 0.84 MeV ^7Li ions. The alpha particles (or ^7Li ions) interact with the gas on one side of the chamber while the gamma-ray background produces ionization on both sides of the chamber. The net current flow from the middle electrode is due only to the neutron ^{10}B interactions. This type of chamber can operate over a large range of neutron fluxes and is part of the instrumentation of most nuclear reactor systems.

Another widely used ionization chamber is the large area fission chamber shown in Figure 3.7. Neutron interaction with the uranium coating produces two fission fragments with a combined energy of approximately 180 MeV. Only about one half of this energy is

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available for deposition into the gas. Still the fission fragment energy deposition far exceeds gamma-ray energy deposition. This type of ionization chamber is finding wide use in nuclear reactors.

The proton recoil ionization chamber is used to measure fast neutron fluxes external to a nuclear reactor by the (n,p) reaction of the fast neutron with a polyethylene liner in the ionization chamber. Proton recoil chambers are insensitive to thermal neutrons and low levels of gamma ray radiation.

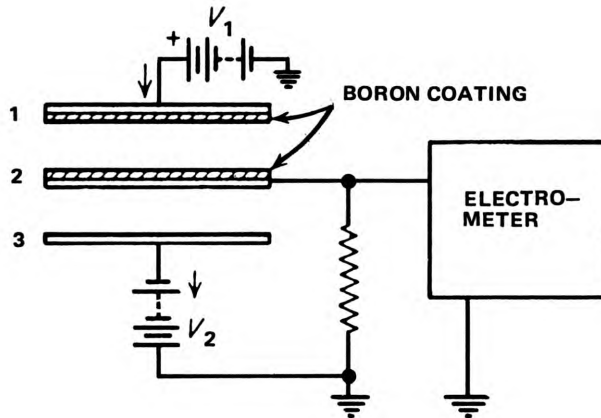


FIGURE 3.6 COMPENSATED ION CHAMBER FOR HIGH-LEVEL DETECTION.

There are a few other Boron coated ionization chambers for nuclear reactor instrumentation that operate both in the pulse and current modes depending on the level of the neutron flux. However, with the exception of the neutron sensitive ion chambers just described, most neutron detectors in common use are of the proportional counter type (Chapter 4).

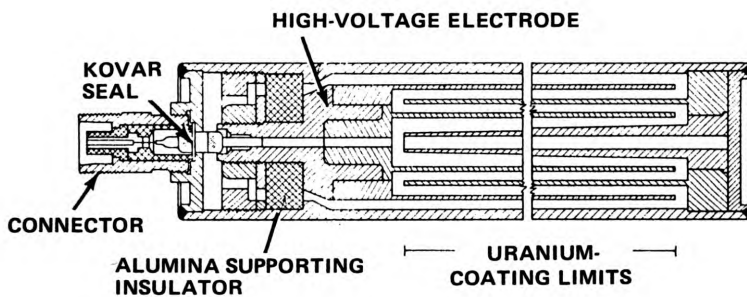


FIGURE 3.7 HIGH EFFICIENCY FISSION CHAMBER.

A special purpose ionization chamber, the extrapolation chamber, has been developed for measuring absorbed dose in any arbitrary material at energies below which most instruments fail to

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satisfy the Bragg-Gray principle (Failla, 1937). This chamber consists of a pair of electrodes made from the material of interest, and arranged in a manner that facilitates variable electrode spacing. The active volume frequently takes the shape of a small coin-shaped region above the central collection electrode located inside a wide guard electrode. The upper electrode is fabricated of a thin metal foil attached to a rigid support ring. Micrometer screws are employed to accurately adjust the electrode spacing. A series of measurements are made with different electrode spacings. The ion current per unit volume is then calculated and extrapolated to the case of zero spacing. With this progressively smaller volume the Bragg-Gray principle (see Section 1.1.4) becomes valid again, and the absorbed dose rate can be calculated from the extrapolated (zero spacing) ion current.

Extrapolation chambers have found use in a large variety of dosimetric problems, including the dosimetry of beta-ray sources (Loevinger, 1953, 1954) and the measurement of dose rate in electron beams (Boag et al., 1951).

A cavity chamber designed by Shonka and Wyckoff in 1964 that is finding increasing use as a calibration standard is shown in Figure 3.8. Air-equivalent conducting plastic is used for the walls and central electrodes. In this design the guard ring is welded to the outer case through an intervening insulating washer. The complete outer electrode and guard ring assembly can be unscrewed from the cable and replaced by a similar assembly with thinner walls for use with low-voltage x rays. Both current and integrating modes of operation are employed.

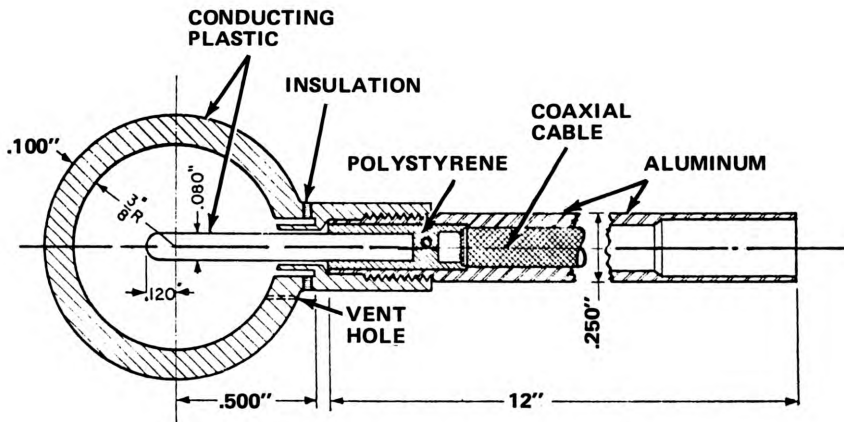


FIGURE 3.8 STANDARD CAVITY (SHONKA-WYCKOFF) IONIZATION CHAMBER.

An early National Bureau of Standards 2π geometry large cavity design is shown in Figure 3.9. This design is used only for the measurement of low energy alpha particles and beta particles from

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radionuclide sources, which are placed in the bottom section of the source mounting ring. This atmospheric pressure spherical design has a narrow cylindrical collector that results in a non-uniform electric field throughout the spherical volume. The consequence of this is that the chamber response is a function of energy and type of radiation to be measured, and the saturation plateau is poorly defined. A more advanced design using cylindrical geometry is shown in Figure 3.10. This combined standard beta-gamma radiation reference chamber is capable of precise placement of the sources to be measured. Beta particle emitters are placed in the lower chamber and gamma-ray emitters (e.g., sealed liquid radionuclide sources) in the center section. The design is typical of cylindrical chambers and provides a large volume uniform electric field resulting in a flat saturation plateau over a large voltage range. These instruments are usually surrounded by appropriate shielding to ensure that external influences, such as electric fields and background radiation, do not affect the measurement.

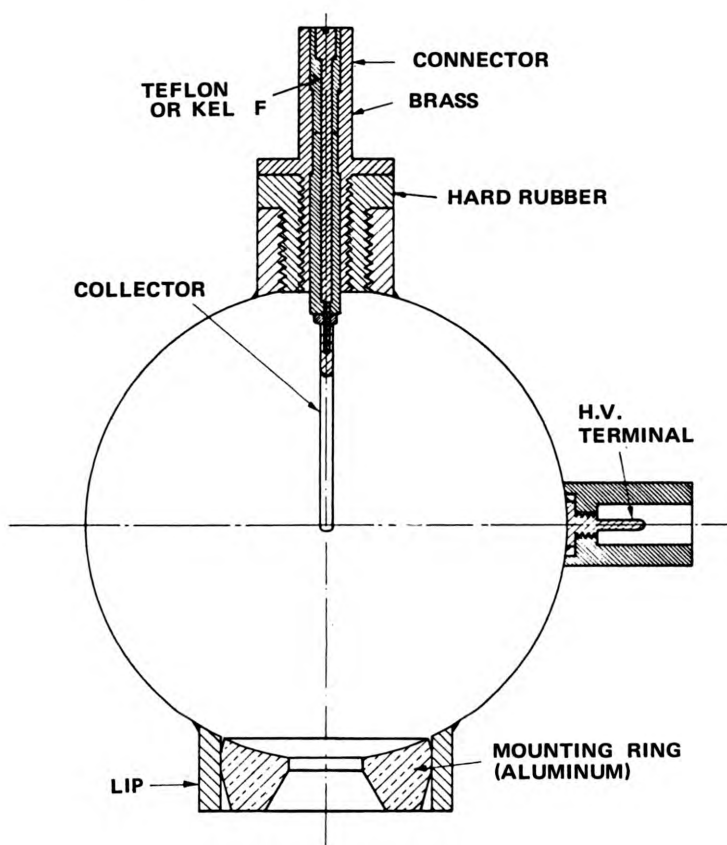


FIGURE 3.9 THE NATIONAL BUREAU OF STANDARDS
 $2\pi\beta$ IONIZATION CHAMBER.

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The most sensitive gamma ray ionization chambers are those designed to operate with high pressure gas. The precision "well" ionization chamber in Figure 3.11 was designed to achieve as close to 4π geometry as possible by allowing placement of a sample ampule in the center of the chamber. There is always the possibility of gas leakage in high-pressure ionization chambers, even though advanced stainless steel, ceramic, or glass technology is used. Therefore, periodic confirmation of chamber calibration using a check source is recommended. High pressure "well" chambers are commonly used in "dose calibrators."

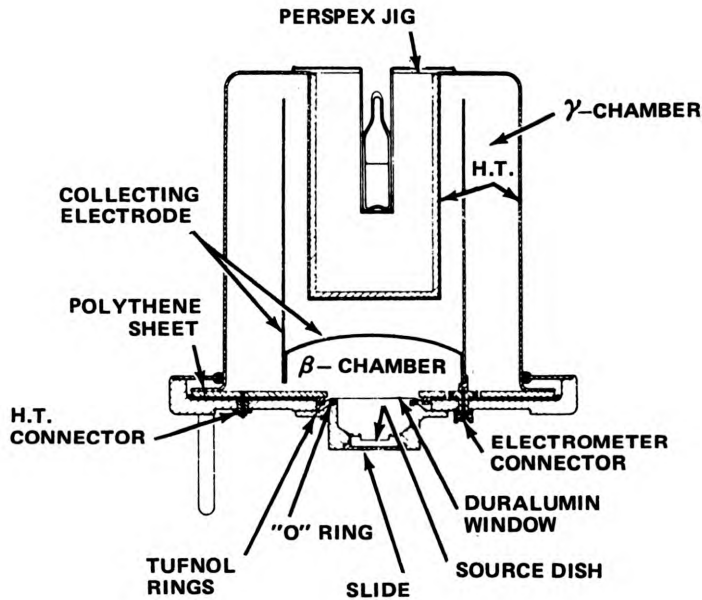


FIGURE 3.10 COMBINED BETA-GAMMA STANDARD IONIZATION CHAMBER FOR LARGE SOURCES AND SMALL AMPOULES.

3.3.4 Integrating Ionization Chambers

As a class, integrating ionization chambers are generally treated at the same time as mean level current instruments. Therefore the same reference sources that were cited for the mean level current instruments are also useful for this category. Mean-level ionization chambers are sometimes operated in the integrating mode. This is accomplished by utilizing the electrical capacitance between two electrodes (i.e., in a condenser) for charge storage, and determining the total charge collected by measuring the change in potential due to irradiation. The simplest form of integrating ionization chamber is the condenser chamber which consists of a

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thimble ionization chamber connected permanently to a condenser. A cross-sectional view of a commonly used commercial instrument, the Victoreen condenser chamber, is shown in Figure 3.12. The thimble chamber in which the radiation produces ionization is located at the right, while the central collecting electrode permanently connected to the condenser is located on the left. The condenser consists of a conducting layer of carbon coated on the inside of a hollow insulator and an outer metal shield surrounding the insulator. The metal shield is connected to the conducting layer of carbon located on the inside surface of the thimble chamber. Figure 3.12(b) shows the equivalent electrical circuit of the condenser-thimble chamber combination with C representing its capacitance, i.e., the total capacitance between the collecting electrode and the surrounding outer sheath. A large part of this capacitance is located in the condenser where the positive and negative charges face each other across a thin polystyrene insulating layer.

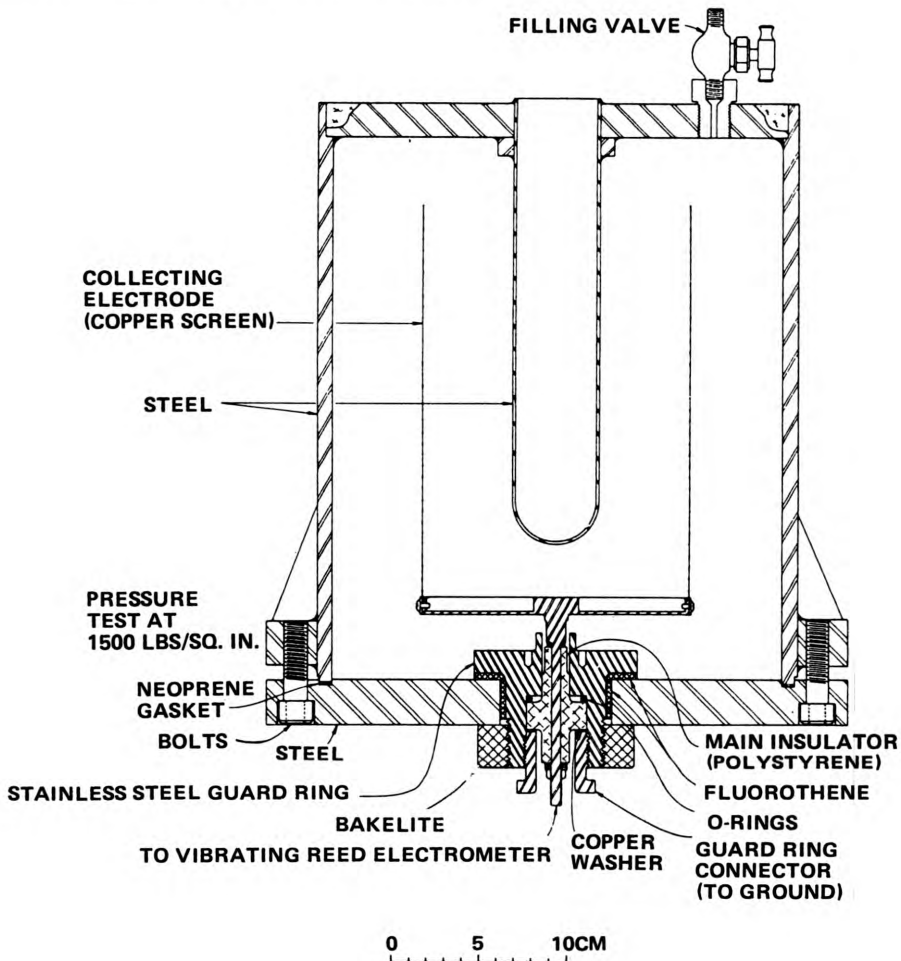


FIGURE 3.11 HIGH PRECISION, HIGH PRESSURE
IONIZATION CHAMBER.

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The condenser chamber sensitivity is determined by the ratio of the air volume to the chamber capacitance. High sensitivity chambers, i.e. those with a large volume and a very small capacitance C . Alternatively, chambers for measuring large exposures must have small volumes and large electrical capacities. Sensitivities are usually determined by measuring the voltage drop of a fully charged condenser for a known exposure. This avoids the necessity of making precise volume and capacitance measurements.

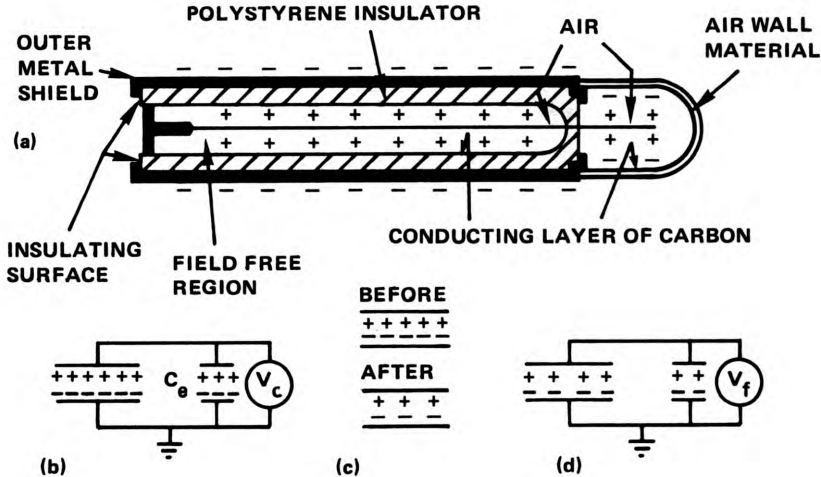


FIGURE 3.12 (a) SCHEMATIC DIAGRAM OF 25R VICTOREEN CONDENSER CHAMBER. (b) CHAMBER OF CAPACITY C CONNECTED TO ELECTROMETER OF CAPACITY C_e FOR CHARGING. SHOWN IN CHARGED STATE. (c) DIAGRAM SHOWING DISCONNECTED CONDENSER WITH 6 CHARGES BEFORE AND 3 CHARGES AFTER IRRADIATION. (d) DIAGRAM SHOWING PARTIALLY DISCHARGED CONDENSER RECONNECTED TO FULLY CHARGED ELECTROMETER. ONE CHARGE HAS BEEN TRANSFERRED TO THE CONDENSER. (JOHNS AND CUNNINGHAM, 1964)

Figure 3.12 (b,c,d) illustrates an exposure measurement made with a charged condenser chamber. The chamber of capacitance C and the electrometer of capacitance C_e are both charged up to a potential V_0 . The condenser chamber is then disconnected from the electrometer and placed in the radiation field being measured. The charge loss that occurs as a result of the ionization produced in the thimble chamber air volume is shown schematically in Figure 3.12(c). Finally, the partially discharged condenser chamber is connected to the fully charged electrometer for the completion of the measurement. A more complete treatment may be found in Johns and Cunningham (1964).

In past years the basic instrument for the measurement of the charge on a condenser chamber has been some type of string

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electrometer. More recently, however, electronic devices such as vibrating reed electrometers, have essentially replaced the string electrometer. An alternated approach is to connect the condenser chamber to an independent voltage source and measuring system. The measuring device may be an integrating system that measures the total charge collected in a given time, or it may be a current measuring system giving a reading proportional to the instantaneous ionization intensity, or more accurately, to the average intensity over a short time interval determined by condenser chamber and measuring system response times. The Baldwin Farmer Substandard X-ray Dosimeter (Farmer, 1955) is an example of this type of application.

The pocket dosimeter is a condenser ionization chamber that is used in virtually all nuclear facilities. It is frequently carried by personnel when performing work where radiation exposure is anticipated. A typical self-reading pocket dosimeter is shown in Figure 3.13. Contrary to what one would expect from the name, this dosimeter measures the discharge of the condenser potential with the scale indicating exposure (Roentgen). Its scale could just as easily be made to read "Rads" in a particular instrument.

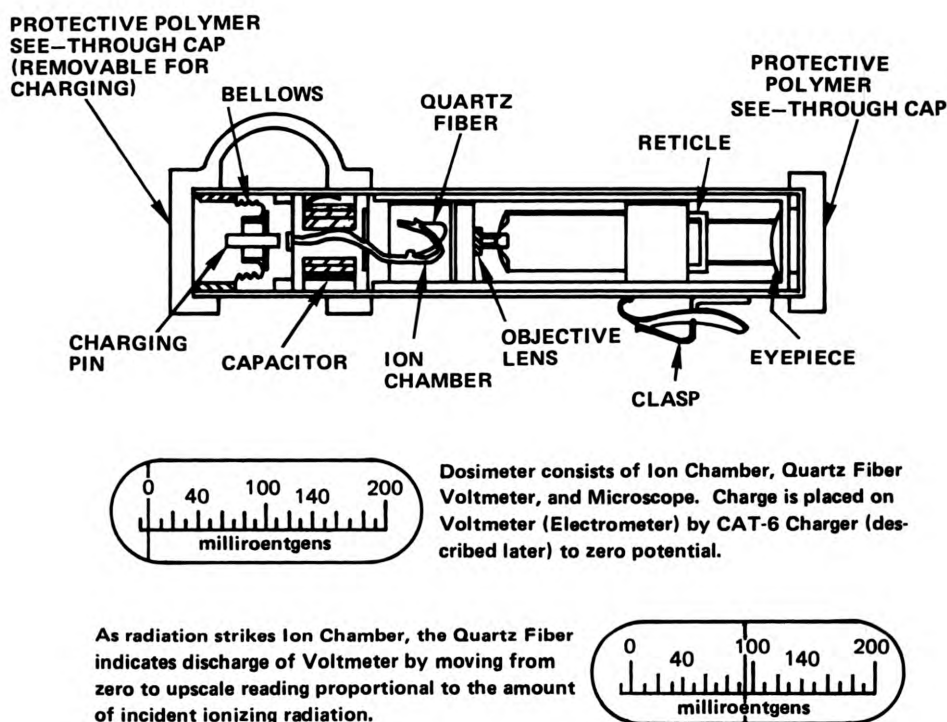


FIGURE 3.13 THE SELF READING IONIZATION CHAMBER DOSIMETER.

The basic component of the dosimeter is a sealed ionization chamber containing a quartz fiber electroscop as the indicating element. The image of the quartz fiber is projected on a reticule and

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can be observed through a built-in microscope. After the instrument has been charged by an external charging unit, the charge is distributed over the quartz fiber and the wire electrode on which the fiber is mounted. The charge causes the fiber to deflect away from the electrode. The amount of deflection is dependent on the amount of charge the instrument has received. The adjustable output of the charging unit allows the operator to set the fiber to a point corresponding to zero on the reticule scale.

When the air in the ionization chamber is ionized, the resulting electrons are collected by the positive charge electrode, reducing its charge and consequently producing a decrease in the deflection of the fiber. This decrease is directly related to the quantity of radiation producing the ionization. The position of the fiber on a scale marked in Roentgens indicates the integrated exposure for x-ray and gamma-ray radiation.

Many other types of condenser chamber dosimeters are available with ranges of 0-200 mR to 0-1000 R. Energy response can be tailored by using appropriate construction materials and chamber design. The addition of a boron coating on the chamber wall makes the condenser chamber capable of measuring thermal neutrons as well. A shortcoming of some direct reading dosimeters is that they are shock sensitive so that readings may change if the dosimeter is dropped. Some condenser chambers do not hold their charge over multiday periods. The resulting error, however, is on the conservative (overdose) side, and depends on chamber design and the cleanliness of the insulator terminals. The following specifications (ANSI N13.5, ANSI N322) should be considered as the minimal acceptable for meaningful measurements:

1. Energy response: $\pm 10\%$ (response/Roentgens) over its specified range.
2. Recombination loss: 10% at maximum exposure rate specified.
3. Self discharge: 2% full scale per day.
4. Measurement error: $\pm 12\%$ (2σ) excluding direction of response after environmental corrections, if required.
5. Geotropism: $\pm 5\%$ maximum scale value.
6. Angular response: Greater than 70% of maximum response to 50% angular deviation position.

3.3.5 Fission Counters as Neutron Detectors

Fission counters operating in the ion chamber region are constructed by placing coatings of ^{235}U , ^{233}U , or ^{239}Pu on the chamber walls. The fission cross sections of these nuclei for thermal neutrons are much smaller than those for ^{10}B or ^3He . Therefore counters based on the fission reaction are much less sensitive. However they have a much greater gamma field discrimination capability. They can operate in gamma fields of up to

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10 R/hr. The general characteristics of fission chambers and other ionization chamber type neutron measurement systems are treated in monographs by Rossi and Straub (1949), Price (1964), Beckurst and Wirtz (1964), Marion and Fowler (1960), and Knoll (1979). Their performance characteristics for radiation protection applications are set forth in special reports such as ICRU 20, 1971 and NCRP 57, 1978. Figure 3.14 shows a series of integral pulse height

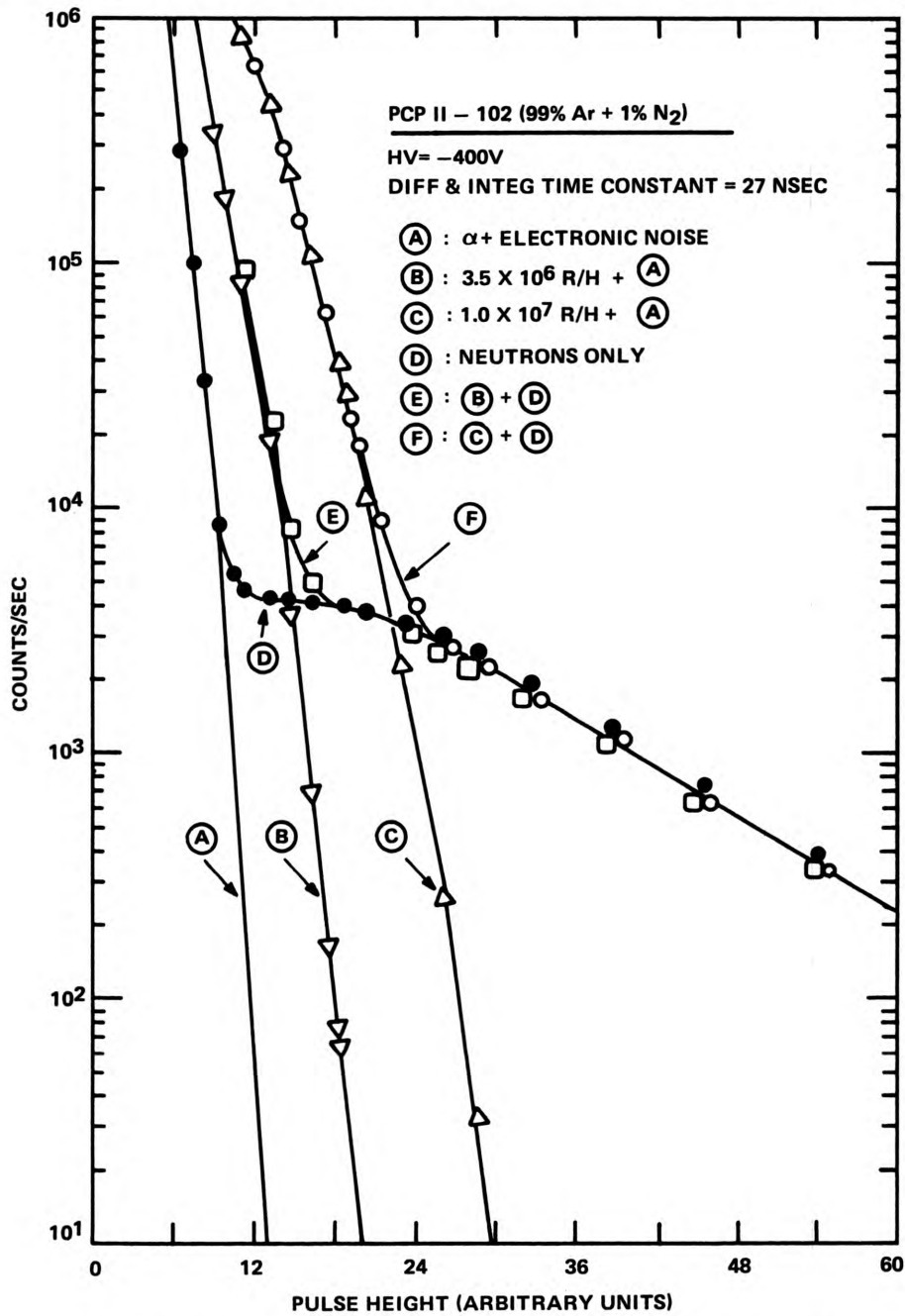


FIGURE 3.14 DETECTOR RESPONSE IN GAMMA AND NEUTRON FIELDS.

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distribution curves for a fission counter filled with a mixture of argon and nitrogen exposed to neutrons and gamma fields of varying intensity (Buhl et al., 1970). This illustrates their relative insensitivity to gamma fields. Even for gamma fields as high as 10^7 R/hr, the fission counters retain much of their neutron sensitivity, even when, as is a common practice, the discriminator level is set to yield a gamma-field-only counting rate of 1 cps.

3.4 SUMMARY

Ionization chambers vary widely in their utilization, mode of operation, geometrical configurations and material composition. They are used for measurement of both particulate (alpha, beta, neutrons and fission fragments) and short-wavelength (x- and gamma-ray) electromagnetic radiations. Their normal use is as portable survey instruments, personnel monitors, dose calibration tools, transfer standards, national roentgen standard, fission counters and reactor power meters and heavy charged particle spectrometers. Their common modes of operation are integrated charge (voltage change), mean level ion current and pulse mode. The absorbed energy dependence of their response make them useful for dose (dose rate) as well as exposure (exposure rate) measurements and spectrometry as well as event counting. Their outputs are scaled in terms of rads (dose), roentgen (exposure) and/or cpm (emission or flux). Their most typical electrode geometries are in the form of either parallel plates or coaxial cylinders of conducting media which are insulated from each other. The usual gas filling for ionization chambers is dry air at atmospheric pressure, but other gases are used for special purposes. Hydrogen, nitrogen and organic gases are used for tissue equivalent chambers, while high pressure Ar or Xe is used for high sensitivity gamma field measurements in the micro R range and Ar-N₂ mixtures are used for fission ion chambers for neutron counting.

Wide range ion chamber survey instruments, which have operating ranges from the mR/hr to thousands of R/hr and usually have dry air as the fill gas, operate in the mean level ion current mode, and have their electrometers (most recently MOSFET) responding to the chamber ion current output to yield a real time dose (or exposure) rate indication in response to the presence of an ionizing radiation field.

Ion chamber dosimeters, such as the pocket dosimeter or more complex types, with ranges of 0-200 mR to 0-1000 R, are again usually dry air filled and operate in the integrated charge mode. Such instruments may use either a simple electroscope or electrometer type charge measurement system and usually are calibrated in terms of exposure or absorbed dose. Observation of the extent of exposure usually has to wait until the end of the measurement. Air wall, air equivalent, tissue equivalent systems usually have similar characteristics and differ only in construction, composition and

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utilization, and usually use more sensitive and accurate electrometers, whose outputs are calibrated in terms of absorbed dose.

Special ion chamber systems which operate with internal sources operate in the mean level ion current mode and again use sensitive electrometers. The above comments are also true for high pressure ion chambers for wide range and low level (micro R/hr) gamma field measurement systems.

Ionization chambers equipped with fissioning neutron sensitive material, such as U-235, acting as charged particle radiators, may be operated either in the mean level ionization current mode or pulse signal mode. However, for other neutron sensitive radiator materials, such as B-10 (for thermal neutron sensitivity) or hydrogenous proton recoil media (for fast neutron sensitivity), only the mean level ion current mode of operation can be used. Such ion chamber systems provide truly wide range performance capabilities of 6 to 10 decades of neutron flux/reactor power measurement, with sensitivities of approximately 10^{-14} amp/unit neutron flux or 10 cpm/unit neutron flux. NBS has developed a system of fission threshold foils for use in standard fission pulse ion chambers for absolute neutron spectrum measurements.

The simplicity of concept, generality of response and diversity of configurations have made the ionization chamber the most widely used and universally applicable radiation measurement system available.

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CHAPTER 4

PROPORTIONAL COUNTERS

W. H. ELLIS

4.1 INTRODUCTION

The proportional counter derives its name from the fact that the amplitude of the output pulse is proportional to the initial ionization. These detectors, which can be considered ionization chambers with internal signal amplification, are operated at voltages well within the gas multiplication region, but sufficiently below the Geiger-Mueller region threshold to ensure proportionality of their output pulses. Gas amplification factors of from 10 to 10^4 are obtained without loss of this proportionality; a great advantage compared to the stringent amplification requirements for such chambers when operated within the ionization chamber region. Many of the general references cited in Chapter 3 for ionization chambers such as Rossi and Staub (1949), Curran and Cragg (1949), Wilkinson (1950), Frazen and Cochran (1962), Price (1964), Sharpe (1964), Fenyves and Haiman (1969), and Knoll (1979) are also applicable here.

Proportional counters can be used for measurement of both charged particles and photons. They are also quite useful for measurement of neutrons. A wide range of fill-gas compositions are in use. The major limitation of these counters is associated with the electron affinity and electron attachment characteristics of the fill-gas constituents, therefore, fill-gases exhibiting electron attachment have to be avoided. Thus, the content of oxygen and other electronegative (electron-attaching) species must be kept to a minimum in the fill gases used. Noble gases, either pure or as constituents in mixtures are useful proportional counter gases for gas multiplication factors below ~ 100 . Beyond this point, a polyatomic additive to the gas is needed to stabilize the discharge. This addition reduces photon ionization, which can cause deviation from proportionality. Argon is the most widely used noble gas for proportional counters. A 90% Ar - 10% CH_4 mixture is known as P-10 gas and is the most commonly employed general purpose gas. Except for helium which does not have large stopping power, the other noble gases are used for applications which require gases with large stopping powers. Hydrocarbon gases, such as ethylene and propane, are used where stopping power is not a factor. Hydrogen is used (for proton recoil counters) where stopping power is not a factor. Sealed

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tubes as well as open tubes (continuous flow gas-filled systems) are utilized for proportional counters. Sealed systems usually feature a coaxial design consisting of a cylindrical metal shell (cathode) and a fine central wire (anode). The small radius of the wire provides for a large electrical field strength near the anode surface. The anode may also have the form of a wire loop. In this case the cathode has the shape of a hemisphere. This configuration is commonly used for gas-flow systems. Stainless steel, nickel or chrome plated brass are widely used for electrodes. Nylon, polyethylene, and other polymeric organics are popular structural materials especially in the construction of tissue equivalent chambers.

Since the charge collected in proportional counters is proportional to the number of primary electron-ion pairs formed, they can be used, in principle and in fact, for measurement of absorbed dose. Certain proportional counters which have tissue equivalent fill gas and wall compositions can be used to measure tissue absorbed dose directly. Achievable sensitivities range in the order of 10^3 to 10^4 times that of ionization chambers (of the same volume and gas fill).

Today's semiconductor counters have largely replaced gas proportional counters for alpha-particle spectrometry. Also, lithium drifted silicon diode detectors are gaining more and more acceptance for applications involving the spectrometry of low energy x-ray sources, especially for energies around a few keV. However, for straight alpha and beta particle counting, as well as fluence and fluence rate measurements, gas-flow proportional counters are still the most widely used systems (NCRP 58, 1978). Their popularity is due to the fact that the proportional response of the counter allows discrimination between alpha and beta particle events. The use of windowless or almost windowless configurations with gas proportional counters minimize alpha and low-energy beta particle losses or energy degradation by the window material. High count rates are achievable, and due to the short resolving times of these counters, no or only very small dead time corrections are required.

Another application of proportional counters is neutron counting. Systems are available for detection and measurement of each of the three principle neutron energy groups (Knoll (1979), Beckurst and Wirtz (1964), Marion and Fowler (1960), Kiefer and Maushart (1972), and Lorenz (1972)). Proportional counters may be used to measure thermal neutron flux densities and fluence. Absorbed dose or dose equivalent for fast neutrons can be measured over limited energy ranges. Thermal neutron detection is accomplished through use of the $^3\text{He}(n,p)^3\text{H}$ and the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reactions. Fast neutron measurement is accomplished by utilization of the proton recoil process where fast neutrons are elastically scattered on the hydrogen nuclei transferring a significant amount of kinetic energy to these hydrogen nuclei. When used in combination with moderators, thermal neutron detectors can be applied to measuring the fluence of fast neutron beams over a wide range of energies from approximately

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0.01 to 20 MeV. A relatively flat energy response is achieved over a wide range which earned this system the name "long" counters.

Gas flow proportional counters are also used to a limited extent as special purpose high sensitivity portable survey meters for low level alpha, low energy beta (tritium) detection and contamination monitoring. Likewise, internal gas proportional detection systems are applied for counting pure beta emitters and for x-ray counting of radionuclei exhibiting electron capture.

4.2 PRINCIPLES OF OPERATION

4.2.1 Gas Multiplication Process

The most representative geometry for proportional counters is a cylindrical design with the outer shell (cathode) coaxially arranged with respect to a central electrode (anode). For a counter with cylindrical geometry the electric field strength $E(r)$ is given by:

$$E(r) = \frac{V_0}{r \ln(b/a)} \quad (4.1)$$

where r is the radial displacement from the axis, a and b are the inner and outer electrode radii, and V_0 is the applied voltage. The inner electrode (anode) usually a wire, ranges from 0.001 to 0.010 inches (0.025 to 0.250 mm) in diameter. Thus the electric field strength $E(r)$ is large and increases sharply close to the anode surface.

Therefore, when increasing the voltage (V_0) between the electrodes, eventually a point will be reached at which the electrons drifting near the anode will gain between collisions sufficient energy from the applied electric field to cause secondary ionization of the gas atoms or molecules. These secondary electrons will have sufficient energy to cause further ionization by collision with gas atoms. The result is gas amplification. If the applied voltage is further increased, the field strength required for ionization by electron collision extends farther out into the gas volume, increasing the number of secondary ionization events. This results in a gain which may be several orders of magnitude. This expanding cascade of secondary ionization is known as "Townsend Avalanche". It is generally confined to a region extending only a fraction of a millimeter from the anode surface. Even so, within this region, a gain of up to 10^4 can take place. The gas multiplication process is described in detail in several excellent monographs on gas filled counters, i.e., Rossi and Staub (1949), Curran and Cragg (1949), Wilkinson (1950), and Fenyves and Harmon (1969).

This phenomenon is illustrated in Figure 4.1 for alpha and beta particles using P-10 proportional counting gas (Price, 1964). Also

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shown for comparison is the characteristics exhibited by the same source and gas-flow counter system operated with a Geiger-Mueller counting gas. As can be seen, the gas amplification for the latter gas mixture is less stable, rising rapidly to reach the GM threshold at relatively low voltages, while the P-10 gas continues to exhibit proportionate differences between alpha and beta responses.

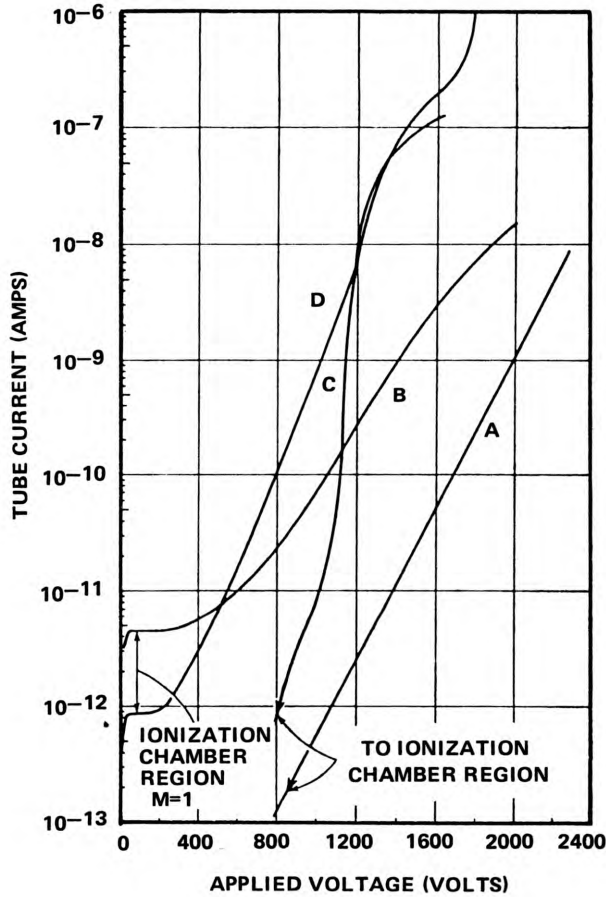


FIGURE 4.1 RELATIVE GAS AMPLIFICATION IN "P-10" GAS (10% METHANE - 90% ARGON) FOR α PARTICLES (CURVE A) AND β PARTICLES (CURVE B) AND "GEIGER-MUELLER" GAS (1.3% ISOBUTANE - 98.7% HELIUM) FOR α PARTICLES (CURVE C) AND β PARTICLES (CURVE D) (PRICE, 1964).

During the avalanche process, in addition to ionization by electron impact, excitation of gas atoms or molecules also takes place. The excited atoms and molecules do not contribute to the avalanche; however, they de-excite by emitting visible or ultraviolet radiation which may ionize atoms and molecules elsewhere in the tube volume. While such characteristics are important for GM chamber

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operation they are generally undesirable for proportional counters since they lead to loss in proportionality and may cause spurious pulses.

Addition of small amounts of polyatomic gases, such as methane (CH_4), to the fill-gas will suppress these photon-induced effects.

4.2.2 General Proportional Counter Characteristics

The following simple model will serve to illustrate the events taking place in a proportional counter (NCRP 58, 1978). Assume a gas fill of argon. Let m be the average electron multiplication in the avalanche, meaning that each one of n electrons produced in the primary ionizing event gives rise (during its acceleration toward the anode wire) to an average total of $\bar{m}$ electrons. Inelastic collision between the accelerated electrons and gas atoms will cause excitation resulting in emission of photons some of which produce photoelectrons (mainly at the wall). If P_e is the probability for such a photoelectron to be formed per electron, then n electrons generated by the primary ionizing event will produce in the avalanche (in the high-intensity electric field near the wire) an average of $n\bar{m}$ electrons by electron impact ionization and $n\bar{m}P_e$ photoelectrons. The electrons are accelerated towards the anode, producing $n\bar{m}^2P_e$ more electrons by collision and $n\bar{m}^3P_e$ more photoelectrons, and so on.

Thus the final number of electrons (N) collected at the anode wire as a result of each primary event is:

$$N = n\bar{m} + n\bar{m}^2P_e + n\bar{m}^3P_e^2 + \dots \quad (4.2)$$

In a proportional counter, $\bar{m}P_e \gg 1$, multiplication occurs essentially by electron avalanche, therefore the series converges to:

$$N = n \frac{\bar{m}}{1 - \bar{m}P_e} \quad (4.3)$$

For conditions under which the initial assumptions are true $\bar{m}P_e \ll 1$, and $N = n\bar{m}$. Thus the total number of electrons in the avalanche (N) is proportional to the number of primary electron-ion pairs, (n) which makes it a proportional counter. It must be emphasized that m is an average number and the actual number of electrons formed from one generation to the next will fluctuate statistically. $M = N/n$ is called the multiplication factor. Although multiplication factors can be calculated, experimental values for a particular chamber design are much easier to obtain and generally follow the shape of the curves of Figure 4.2 (Rossi and Staub, 1949).

The general characteristics of proportional counters can be described in terms of the diagram presented in Figure 4.3. The

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pulses obtained are amplified and shaped through a preamplifier (PA) and linear amplifier (LA) and then applied to the input of a single channel analyzer (SCA), which for simple systems may be a discriminator-trigger circuit at the input of a scaler unit. For pulses having an amplitude which exceeds the trigger level the discriminator will trigger and generate a logic pulse which is used to drive the scaler and to register a count in the decade unit.

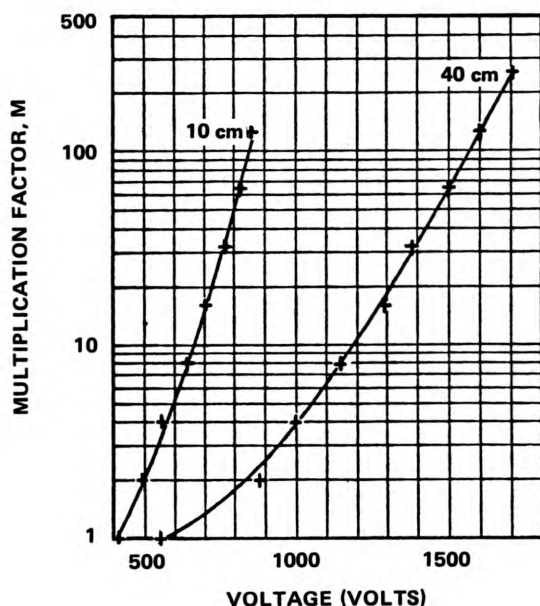


FIGURE 4.2 MULTIPLICATION FACTOR, M , AS A FUNCTION OF APPLIED VOLTAGE FOR ARGON AT PRESSURES OF 10 AND 40 cm OF Hg. RADIUS OF CENTER WIRE IS 0.013 cm, RADIUS OF OUTER ELECTRODE IS 1.105 cm. (ROSSI AND STAUB, 1949).

For low values of the applied voltage, the output pulses will be too small, and fall below the discriminator trigger level, therefore, no logic pulse for driving the scaler will be generated. However, as the applied voltage is gradually increased, gas amplification will cause the amplitude of the proportional counter output pulses to increase until the most highly ionizing events will produce pulse amplitudes which will just exceed the discriminator trigger level, and so generate logic pulses. As the voltage level is increased further, the observed counting rate will increase with voltage until eventually all ionizing events, even those having the lowest ionization capacity will be above the discriminator trigger threshold. When this occurs, the counting rate is observed to reach a "plateau," which means the counting rate no longer changes as a function of voltage until the region of continuous discharge is reached. Any increase in the effective sensitive volume of the

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counter, or the onset of spurious pulses, can cause the plateau to have a positive slope. Low gas gains (low applied voltage) can be selected if modern low-noise amplifiers are used to avoid spurious pulses.

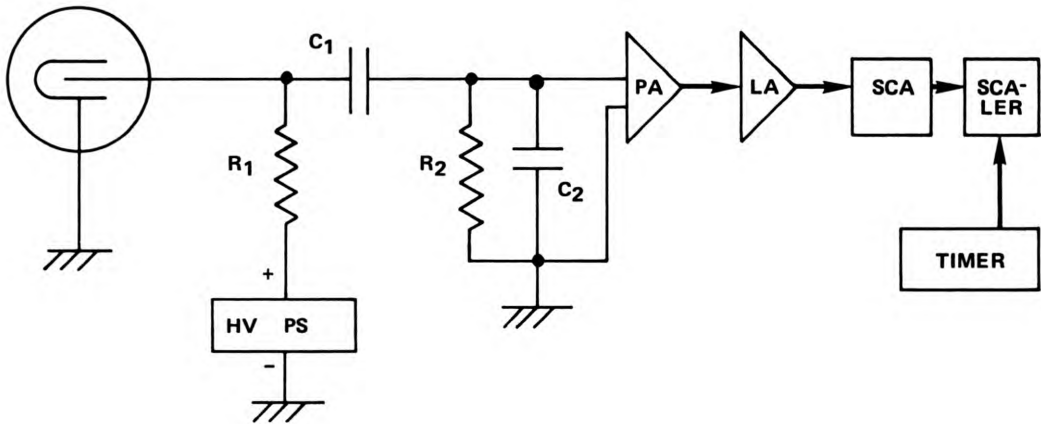


FIGURE 4.3 GENERALIZED PROPORTIONALIZED COUNTING SYSTEM

In the plateau region the pulse height is a function of applied voltage but the number of counts is not. This allows differentiation between radiation sources having different energies. For example, when attempting to determine the alpha contamination on a wipe sample, it is necessary to differentiate against gammas and betas. Alpha energies are normally in excess of a few MeV, beta energies less than a few MeV, and gamma energies (capable of significant deposition) less than 100 keV. The substantial difference in these energy characteristics has the effect that more than one plateau exists. For a fixed trigger level the voltage plateaus for each of these different sources are indeed far apart. If the voltage is set for the alpha plateau, discrimination against beta and gamma energies is almost complete. With the voltage set for the beta plateau, both beta and alpha ionization events are counted and only gamma photons are discriminated against. The points at which the counting threshold and plateau occurs for a given radiation source are functions of the value of initial ionization in the chamber (particle dependent), the amount of gas multiplication (gas-fill characteristics, and high voltage setting), and the level of the discriminator trigger. Preamp and main amplifier gain settings will also contribute to the pulse height at the input of the discriminator. If highly specific alpha counting is desired, one raises the threshold to discriminate first against electronic noise, then against betas and gamma ray interactions.

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After differentiation between alpha and beta counts is accomplished in this way, calibration of the system for alpha and beta detection efficiencies allows the activity of these two types of radiation to be determined.

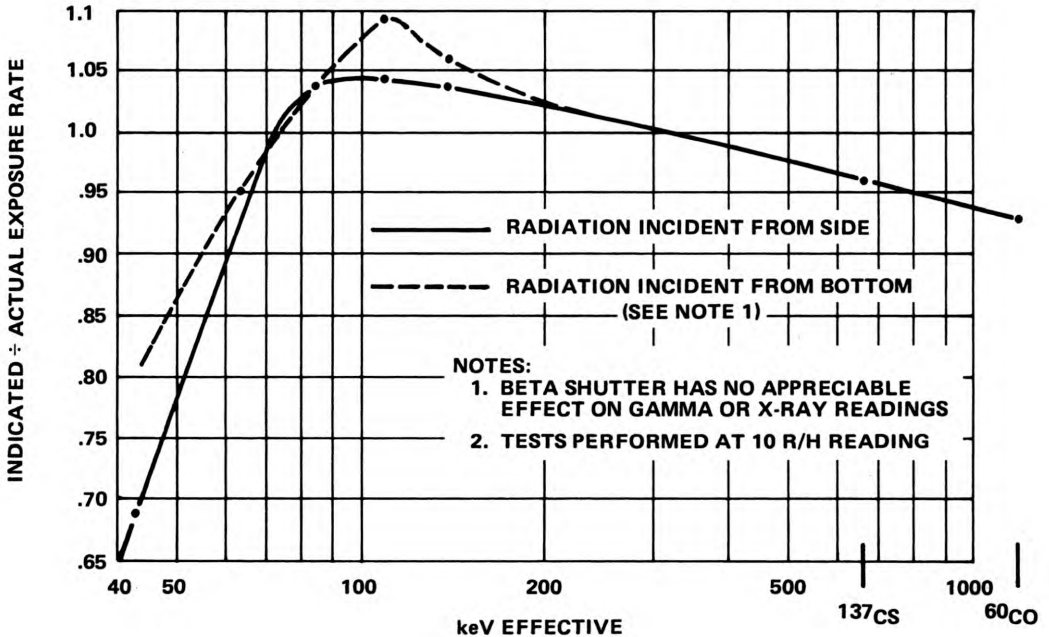


FIGURE 4.4 ENERGY RESPONSE CURVES FOR PROPORTIONAL REGION SURVEY METERS.

Sometimes the radiation source is located inside the counter to avoid window absorption. In this case the counter is operated at atmospheric pressure, generally with methane or methane-argon mixtures flowing continuously through the chamber (flow counter). Output pulses from a beta-particle spectrum vary in amplitude by a factor of 1,000 or more if the counter is operated at elevated pressure. Consequently the associated amplifiers must have good overload characteristics. This can be achieved, for example, with a low-noise charge-sensitive preamplifier and a doubly differentiating main amplifier.

4.3 PROPORTIONAL COUNTER APPLICATIONS

There are three general fields of applications for proportional counters: (1) survey (gamma, beta, and neutron), (2) spectrometry (alpha, beta, gamma, and neutron), and (3) absolute counting (variable geometries, all sources).

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4.3.1 Proportional Counter Survey Instruments

Survey instruments which utilize the proportional region for measurement of gamma and beta exposure are not common, but they offer certain advantages since these instruments do not count, but measure current. In the current mode saturation is no problem. The achievable high sensitivity allows it to observe and amplify a single particle deposition, and therefore the useful range is very large. Pressurized chambers are operated in the so-called amplified ionization chamber region. It is customary to use logarithmic output displays from 1 mR/hr to 1 R/hr or 1000 R/hr. This can be done since their efficiency is quite uniform over a wide range of energies. A typical energy response curve is shown in Figure 4.4.

Survey instruments having large flat plate gas flow probes are more common. They are used for the measurement of alpha and beta contamination on surfaces. These chambers are very efficient (greater than 50% in some cases) and are extremely useful in facilities where radionuclide chemical or biological studies are performed. They must be operated on a flat surface in order to maintain a gas flow blanket, and they usually contain several long center electrodes. A disadvantage is that a supply of gas is usually required. The electronics used in such instruments are not extensive and may be identical to the ones used in portable scintillation and GM counters. Figure 4.5 is a plot of alpha and beta response in a

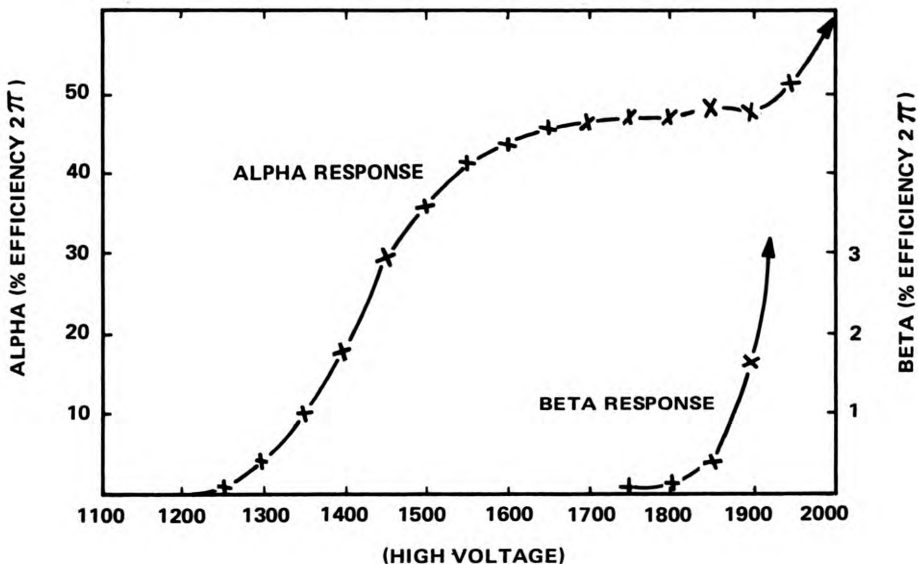


FIGURE 4.5 ALPHA AND BETA COUNTING EFFICIENCY CURVES FOR PROPORTIONAL COUNTERS.

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typical air proportional system (note the obvious discrimination capability). Windowless systems are especially useful in counting low energy beta emitters (tritium, ^{14}C) in the absence of other radionuclide sources.

4.3.2 Laboratory Proportional Counter Systems

The single most widely used type of proportional counter is the windowless gas flow, 2π (geometry) system (Price (1964), NCRP 58, 1978; Emery (1966), and Knoll (1979)). It is used for intermediate and low level alpha and beta sources. Figure 4.6 illustrates the principle elements in this class of gas flow proportional counters. The sample is enclosed within the active volume of the chamber.

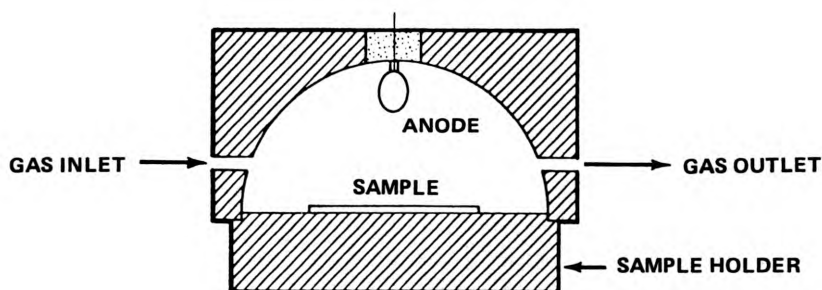


FIGURE 4.6 DIAGRAM OF A 2π GAS FLOW PROPORTIONAL COUNTER WITH A LOOP ANODE WIRE AND HEMISPHERICAL VOLUME.

Following a certain period of purging the counting gas flows continually through the active volume of the counter. The anode has the form of a wire loop. The hemispherical chamber walls and the sample holder constitute the cathode. Figure 4.7 shows a typical response curve for a mixed alpha-beta source. It illustrates the substantial difference between the alpha and beta thresholds, which is used for differentiating between these two types of radioactivity. In order to measure small amounts of alpha activity in the presence of relatively high levels of beta and gamma activity the counting system is operated at applied voltages falling on the alpha plateau region and below the beta threshold. Subsequent measurement of the counting rate with the applied voltage set within the beta plateau region yields the counting rate due to both the alpha and the beta contributions. A 4π -system used for absolute alpha and beta counting measurements is shown in Figure 4.8.

Large computer controlled systems are available. They automatically change samples, subtract background, count for either a fixed or a statistically relevant time, change voltage and time thresholds for alpha and beta detection, etc. An analysis of 100 or

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more samples can be performed automatically this way. Proportional counters which monitor radionuclides in gas samples are also available. These detectors commonly work on a sample-and-mix basis, since in a continuous flow of pure air it is difficult to establish a threshold for beta particles.

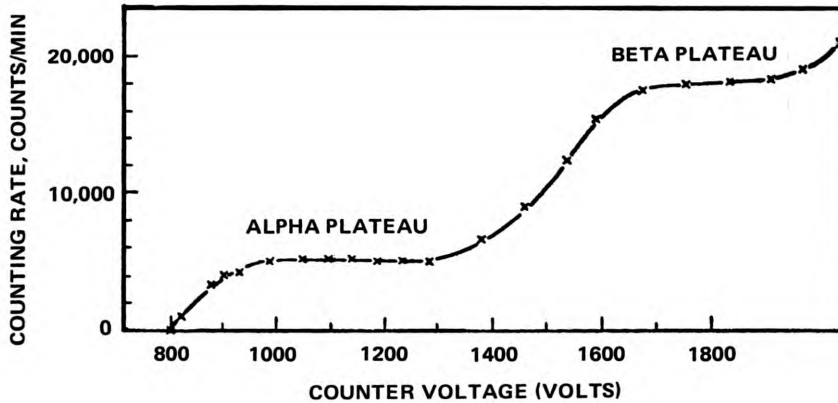


FIGURE 4.7 COUNTING RATE VERSUS COUNTER VOLTAGE FOR A FLOW-TYPE PROPORTIONAL COUNTER WITH A NATIONAL BUREAU OF STANDARDS RA-226 SOURCE IN EQUILIBRIUM (PRICE, 1964).

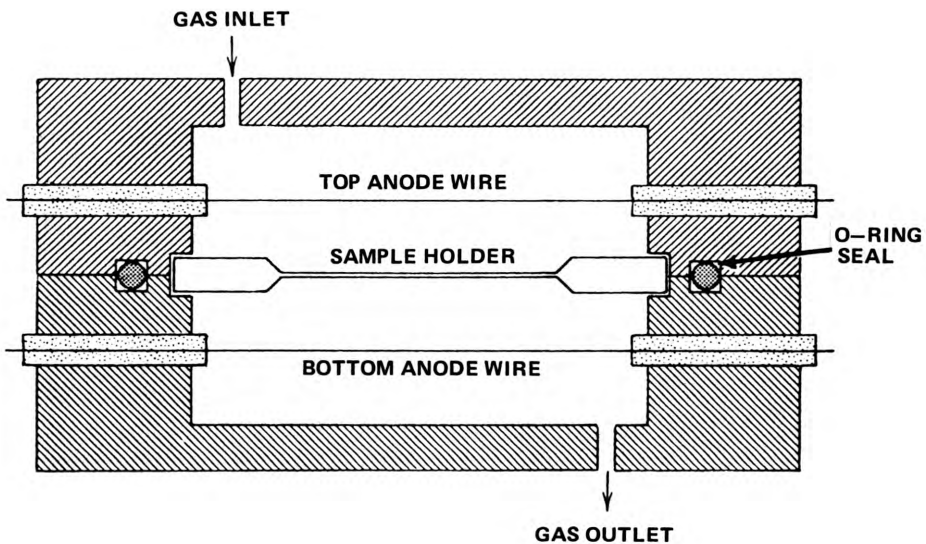


FIGURE 4.8 A 4π GAS FLOW PROPORTIONAL COUNTER USED TO DETECT RADIATIONS THAT EMERGE FROM BOTH SURFACES OF THE SAMPLE. THE TOP AND BOTTOM HALVES ARE PROVIDED WITH SEPARATE ANODE WIRES AND CAN BE SEPARATED TO INTRODUCE THE SAMPLE THAT IS MOUNTED BETWEEN THEM (KNOLL, 1979).

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4.4 PROPORTIONAL COUNTERS AS NEUTRON DETECTORS

4.4.1 BF_3 Counters as Neutron Detectors

Almost all neutron proportional counter systems measure "counts," relating to either neutron flux, spectrally resolved or unresolved, or dose equivalent (rem). The most widely used thermal neutron detector is the BF_3 -filled gas proportional counter which uses the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction. Excellent coverage of the characteristics and applications of BF_3 counters as neutron detectors is given in monographs such as Rossi and Staub (1949), Beckurst and Wirtz (1964), Price (1964), and Knoll (1979). This detector is available in a large variety of sizes, shapes, and sensitivities suitable for a large number of applications. A typical cylindrical chamber is shown in Figure 4.9. The thermal cross

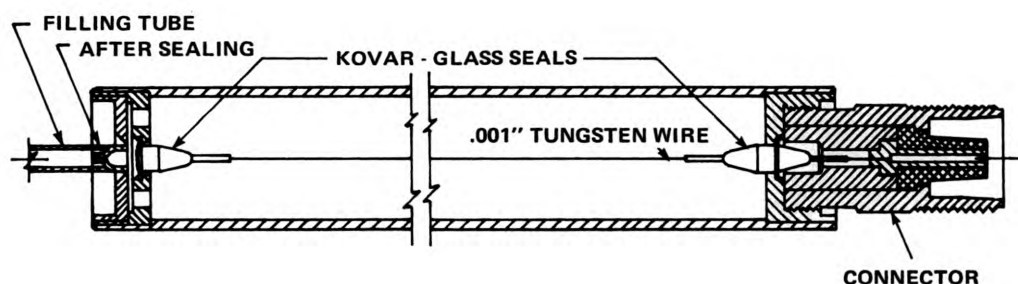


FIGURE 4.9 TYPICAL BF_3 COUNTING TUBE.

section for ^{10}B is quite large (3800 barns at 0.025 eV) and the stopping power of the gas (BF_3) is adequate to achieve almost complete stopping of all of the alpha particles and lithium ions released in the chamber. For proportional counters the BF_3 is usually isotopically enriched to 96% in ^{10}B in order to increase its neutron sensitivity. (The natural abundance for boron is 80% ^{11}B and 20% ^{10}B .) BF_3 is self-quenching since its optical emission occurs only in the IR. It exhibits stable gas amplification and flat counting rate plateau curves. The pulse height spectra expected for complete stopping of the reaction products is shown in Figure 4.10 (a) while Figure 4.10 (b) displays an additional continuum which is caused by partial loss of reaction product energy to the chamber walls in the case of smaller tubes. Partial or complete loss of the energy of either of the two reaction products will result in a continuum, one for each of the particles, the lower being due to alpha wall loss effects and the higher due to effects of lithium ion wall loss. These figures also demonstrate that even with

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wall effects present the pulse heights due to neutron events are generally well above those due to electronic noise and gamma ray interactions. These allow pulse height discrimination without undue loss of neutron sensitivity. However, extremely high gamma fields may result in almost complete loss of the counting plateau as shown in Figure 4.11. The maximum gamma field in which BF_3 counters can be used for thermal neutron flux measurement is approximately 10 R/hr. Also shown in Figure 4.11 is the counting plateau of a boron-lined counter, which in general has inferior plateau characteristics. The reason is that the two product particles formed by the nuclear reaction in the solid must necessarily travel in opposite directions. Therefore only one of them can enter the gas region. Each particle which escapes from the boron lining contributes to a continuum of pulse amplitudes.

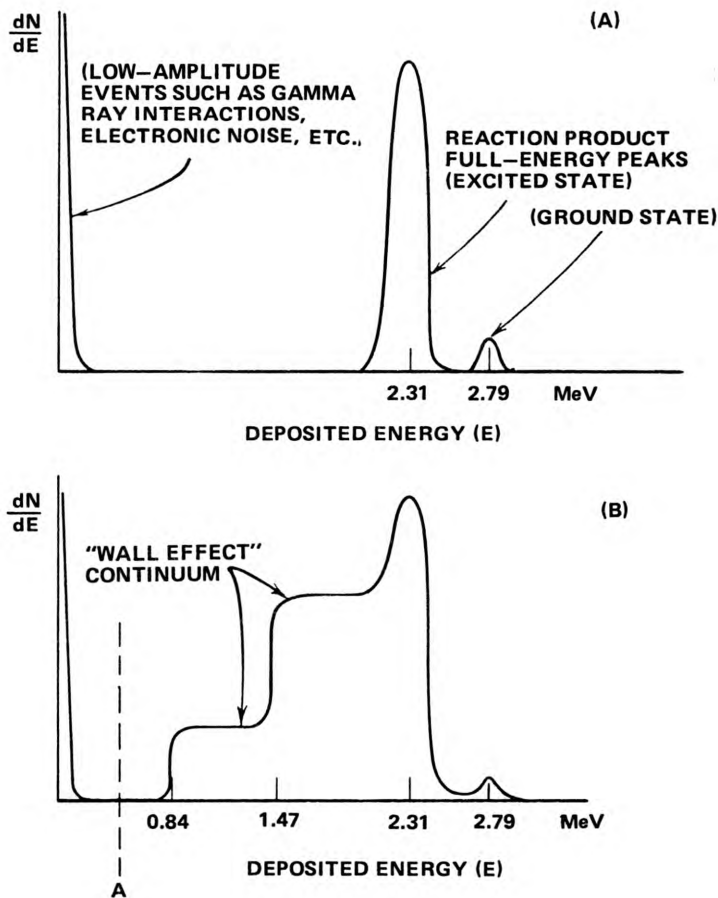


FIGURE 4.10 EXPECTED PULSE HEIGHT SPECTRA FROM BF_3 TUBES. PLOT (A) SHOWS A SPECTRUM FROM A LARGE TUBE IN WHICH ALL REACTION PRODUCTS ARE FULLY ABSORBED, WHEREAS PLOT (B) ILLUSTRATES THE ADDITIONAL CONTINUUM DUE TO THE WALL EFFECT (KNOLL, 1979).

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At high gamma flux densities the finite pulse resolution of the counter may cause gamma pulse pile-up. This is the integration of many of the small gamma pulses that randomly occur into a single pulse large enough to overcome the discrimination threshold. The result is the shortening of the alpha counting plateau as shown in Figure 4.11.

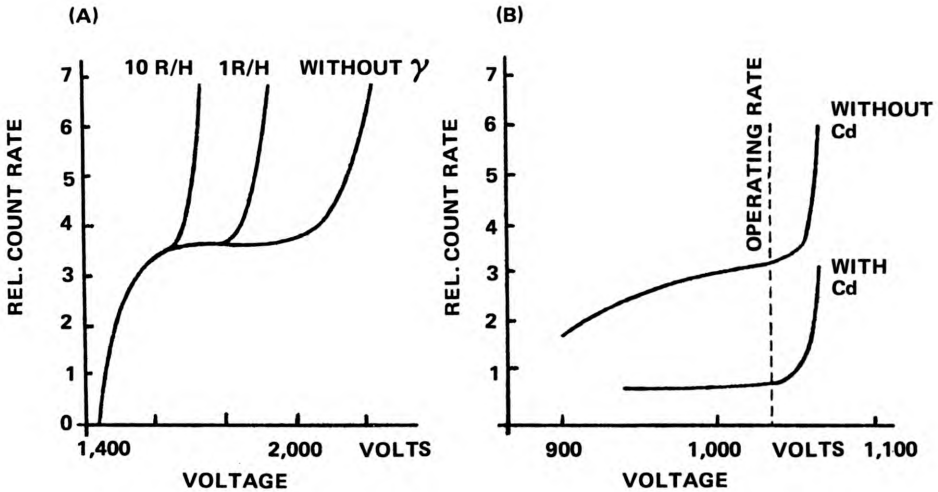


FIGURE 4.11 COUNTER PLATEAU OF (A) A BF_3 COUNTER AND (B) A BORON WALL COUNTER.

BF_3 counters can also be used for measurement of fast neutron fluxes and dose rates. For this application they are enclosed in a shield consisting of a combination of hydrogenous moderator and thermal neutron absorber in an appropriate configuration (Price (1964), Kiefer and Maushart (1972), and Knoll (1979)). Figure 4.12 shows a typical cylindrical system which is known as a "long counter" owing to the relatively flat neutron energy response over a long range, namely from approximately 10 keV to 5 MeV. This type of counter relies on the moderation of fast neutrons since ^{10}B has a large neutron absorption cross section only at thermal energies. These counters are directional. They consist of a BF_3 counting tube inside a moderator assembly (usually paraffin or polyethylene) which is surrounded by a thermal neutron shield to intercept low energy neutrons coming from the outside. Materials used for the shield are cadmium or B_2O_3 . If a source is placed in the normal counting position on the axis at the open face end of such a detector assembly (at the right in Figure 4.12) the neutrons will penetrate the hydrogenous medium roughly parallel to the axis. Consequently they will be slowed down and detected near the point at which they are thermalized. For a given detector assembly only the characteristic depth of penetration changes with neutron energy, not the effective length traveled by neutrons following their thermalization

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(diffusion length). Bare BF_3 counters are relatively insensitive to fast neutrons due to the $1/v$ dependance of the ^{10}B cross section. For this reason even in detectors with moderators, fast neutrons contribute little to the overall detector response during the penetration and slowing down phase. As pointed out above the depth of penetration is a function of their initial energy, therefore the lower energy neutrons will be detected near the face of the detector.

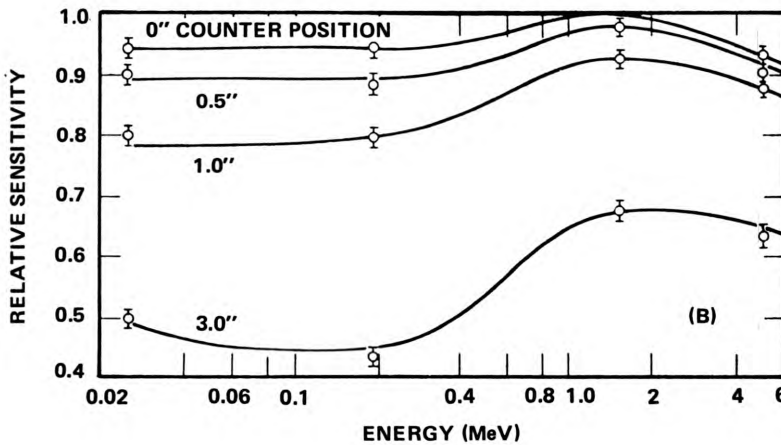
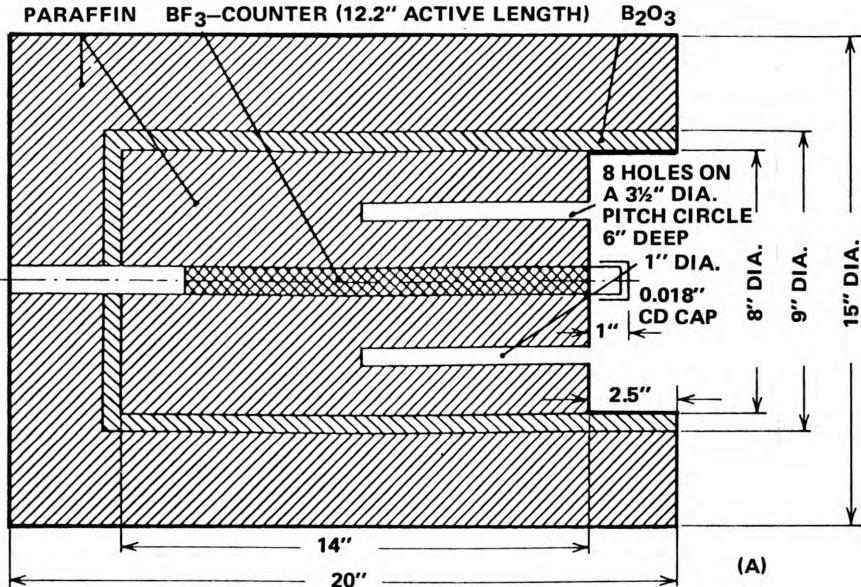


FIGURE 4.12 (A) CROSS - SECTION OF THE LONG COUNTER DEVELOPED BY McTAGGART. (B) RELATIVE SENSITIVITY OF McTAGGART LONG COUNTER VERSUS NEUTRON ENERGY. THE PARAMETER VARIED FOR THE DIFFERENT CURVES IS THE DISTANCE THE END OF THE BF_3 TUBE IS SHIFTED IN FROM THE FRONT OF THE MODERATOR FACE. THE FLATTEST RESPONSE OCCURS WHEN THE TUBE IS FLUSH WITH THE FRONT FACE (BECKURST AND WIRTZ, 1964)

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Higher energy neutrons, however, will penetrate more deeply into the moderator and will be detected further down in the BF_3 tube. Care must be taken that the thermal neutron sensitivity of the BF_3 tube is kept approximately constant along its full length. Consequently the long counter systems exhibit flat neutron energy response over wide ranges of neutron energies.

BF_3 proportional counters encased within cylindrical hydrogenous moderator sleeves are used as dose equivalent instruments (Emery (1966), Kiefer and Maushart (1972)). They provide sensitivities of approximately 2 cps per mrem/hr. They give direct measurements of dose equivalent (H) at the position of maximum dose equivalent (MADE) independently of neutron energy in the range of thermal to 15 MeV. The cylindrical moderator counters are quite directional, particularly at lower neutron energies. On the other hand, BF_3 counters, and especially spherical ^3He counters incorporated in spherical moderators, are much more isotropic in response. However, all counters of this type over-respond by a factor of ~ 4 at a few keV, and under-respond by a factor of ~ 2 at 10 MeV when calibrated at 1 MeV. The fact that neutron fields are generally distributed over quite wide ranges of energy allows the response of such systems to more realistically reflect the actual dose equivalent than indicated for each of the two extremes described above. Under certain circumstances these counters are also suitable for pulsed neutron measurements and are insensitive to gamma fields for exposure rates less than 200 R/hr of ^{60}Co .

Systems which are commercially available feature spherical moderators and provide sensitivities of 1.5 to 15 cps per mrem/hr. They give direct measurement of the MADE independently of neutron energy over a specific energy range and are non-directional. For a single 25 cm diameter sphere, the maximum overestimate of H is four times its actual value at 10 keV if calibrated with neutrons of a few MeV, but it underestimates at neutron energies of approximately 7 MeV. Multisphere systems are also available. They yield information on spectrum shapes and fluence and can be used for accurate H measurements ($\pm 10\%$). Another portable dose equivalent type meter that finds wide use is the so-called Rem Meter. This type of BF_3 counter can be operated either alone or within its thermalizing assembly. It can so span a sensitivity range of neutron energies from thermal to 10 MeV, and the upper limit dose rate is around 5000 mrem/hr. It should be pointed out that "Rem Meter" seems to have become a generic term for any neutron measuring instrument calibrated to read in rem. Thus, the counters discussed in this and the preceding paragraph are all remmeters, as are several other types. Of course, if the BF_3 counter is used alone, without its moderator, it is no longer a remmeter.

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4.4.2 ^3He Counters as Neutron Detectors

Another neutron detection device used is the ^3He gas proportional counter utilizing the $^3\text{He}(n,p)^3\text{H}$ reaction (Price, 1964; Knoll, 1979). ^3He is an acceptable proportional counting gas when combined with a molecular component such as CO_2 or N_2 . However, due to its small atomic number and low density, the stopping power of the ^3He is low for atmospheric gas fill pressures and much of the energy of the proton and triton reaction products is lost to the walls of the chamber. As shown in Figure 4.13, the pulse height spectrum of a typical tube exhibits a "back porch" type structure similar to BF_3 tubes (Knoll, 1979). The pulse height continuum caused by wall effects results in shortened counting plateaus. Reduced discrimination (differentiation) against gamma induced pulses is another consequence. Reduction in the wall effect can be accomplished by several methods: 1) increasing the diameter of the chamber, 2) adding a heavy component such as argon or krypton to the ^3He gas-fill, 3) increasing the ^3He gas pressure to several atmospheres, or 4) a combination of these.

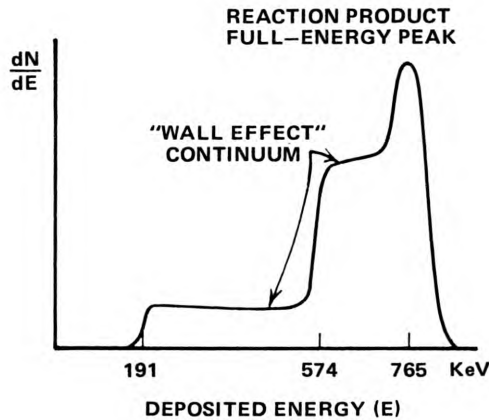


FIGURE 4.13 EXPECTED PULSE HEIGHT SPECTRUM FROM A ^3He TUBE IN WHICH THE WALL EFFECT IS SIGNIFICANT (KNOLL, 1979).

^3He proportional counters can be operated at much higher pressures than BF_3 counters. Due to the higher gas pressure and ^3He atom density the macroscopic cross section and the stopping power increase proportionality, improving both the detection efficiency and wall-loss effects. However, to maintain the same gas amplification and pulse rise time characteristics the chamber voltage has to be increased. Such chambers are preferred for applications where maximum detection efficiency is important.

It should also be noted that high pressure ^3He chambers, as well as high quality ^3He -Ar counters, can have quite a good pulse height resolution, i.e., 2% to 3%. They thus find use as neutron

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spectrometers in the neutron energy range from a few tens of keV to ~1 MeV.

4.4.3 Gas Recoil Proportional Counters

An important method for detecting neutrons in the energy range between 0.1 and 20 MeV utilizes the elastic scattering processes of fast neutrons in hydrogenous substances. The recoil experienced by low mass nuclei (the protons in hydrogenous materials) can be measured using proportional counters, ion chambers, scintillation phosphors, semiconductor devices, and nuclear track emulsions. The energy dependence of the response depends on the design of the detector and its material composition. The proton recoil process is based on the elastic scattering of neutrons. Therefore low mass materials have to be incorporated either into the counting gas or the lining of the chamber walls. The protons formed by the elastic interaction of a fast neutron with an atom share part or all of the neutron kinetic energy and transfers it to the counting gas via ionization and excitation. The elastic scattering cross sections roughly follows a $1/E$ energy dependence above ~10 MeV. However, for n-p scattering the cross section is flat between ~1 eV and ~10 keV, and then falls more slowly than $1/E$ until ~10 MeV. The general systematics and applications of gas recoil processes for radiation detection and measurement is presented in a number of excellent monographs, including Knoll (1979), Emery (1966), Beckurst and Wirtz (1964), Marion and Fowler (1960), Dennes and Loosemore (1961), Kiefer and Maushart (1972), and ICRU 20 (1976).

The simplest proportional counter of this type uses a hydrogen or methane gas filling as the neutron sensitive medium and may have a configuration similar to the chamber shown in Figure 4.14 (Emery, 1966). Detector efficiency is highly energy dependent, hence only relative neutron flux density measurements can be made. For more sophisticated applications, the energy response can be calibrated using monoenergetic neutron sources, e.g. accelerators. For neutron flux measurements involving sources having a distribution of energies, deconvolution of the detector response is required based on response functions obtained from the calibration with monoenergetic neutrons.

Ionization produced in the fill gas by the gamma ray background sets a lower practical limit of approximately 0.1 MeV for the neutron energy which can be measured with proton recoil detectors. However, under low gamma field conditions, or for detector systems for which gamma response can be removed, measurements down to 20 keV are generally not difficult. Measurements down to 2 keV are possible, but they represent something of a "tour-de-force." Even though the system is relatively insensitive to gamma rays, pile-up effects in high gamma radiation fields can influence the recoil detector response. High gamma ray fields also tend to rapidly decompose organic molecular fill-gases such as methane.

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Recoil detectors find considerable use for integral and differential fast neutron energy spectrometry. However, the most important application is radiation dosimetry. Simple counters such as the "Hurst" type, employ polyethylene moderators of various thicknesses and use mixtures of hydrogen, argon, and methane as fill gases. The output response is proportional to the absorbed tissue dose rate over the neutron energy range of 0.2 to 10 MeV. These detectors tend to be somewhat directional in their neutron sensitivity. High neutron flux densities up to dose rates of 100 rad/hr can be measured with less than 5% counting loss with specialized Hurst counters in the presence of gamma radiation. These instruments will tolerate gamma/fast neutron dose ratios of 200:1 and thermal neutron/fast neutron dose rate ratios of 100:1.

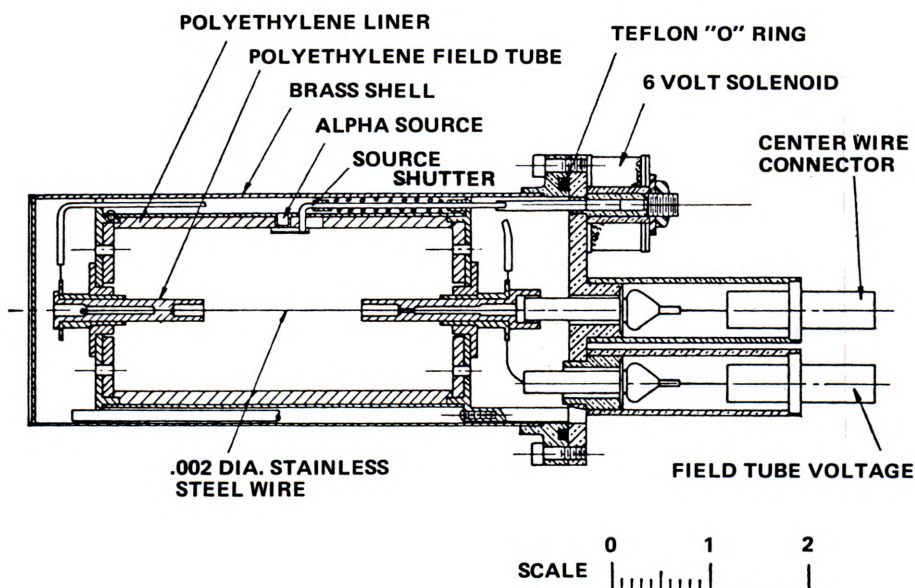


FIGURE 4.14 PROTON RECOIL PROPORTIONAL COUNTER
(HURST & WAGNER, 1961).

The so called absolute proportional counter for fast neutron dosimetry allows direct dose determination by measurement of total ionization which is accomplished by integration over the output pulse, rather than measuring pulse height alone. Figure 4.15 (Emery, 1966) shows integral pulse response characteristics of such a counter for a neutron dose of 1 mrad/hr coming from a Po-Be source. The effects of ^{60}Co gamma rays for several dose levels can be seen. Of course it is possible to discriminate against these gamma rays by raising the discriminator level. However, in doing so much of the detector sensitivity to lower energy neutrons is lost. For this reason correction factors have to be used. For calibration an alpha source is built into the detector. The counter is made of

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polyethylene and has an ethylene gas fill which is tissue equivalent. This is due to the fact that the ratio of the energy absorbed in ethylene to the energy absorbed in tissue is independent of neutron energy at least over the range of 0.1 to 20 MeV. With the threshold set to discriminate against gamma rays, the counter is capable of exact dose determinations only for neutrons above 0.5 MeV.

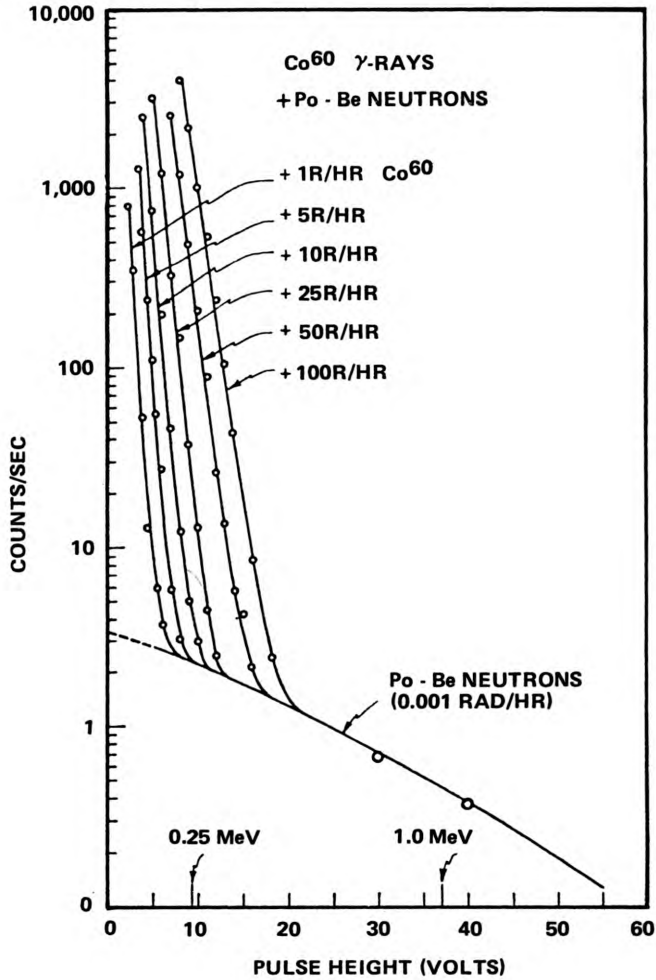


FIGURE 4.15 INTEGRAL PULSE-HEIGHT CURVES FOR MIXTURES OF GAMMA RAYS AND FAST NEUTRONS (HURST & WAGNER, 1961).

The energy sensitivity of the Hurst type meter ranges from 0.2 to 14 MeV, the counting rate sensitivity is approximately 1mRem/hr per count per second and the accuracy is $\pm 10\%$ for Po-Be neutrons. Dennes and Loosemore (1961) describe another recoil counter system which is filled with methane-argon at 20 torr and measures equivalent dose (rem) in the fast neutron region. This system is widely used in Great Britian but is not commercially available. Similar

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instruments have been used for measurements of neutron flux up to 15,000 neutrons per cm^2/sec . They are accurate to within $\pm 70\%$ over the energy range of 0.1 to 14 MeV.

Another type of recoil counter is the Bragg-Gray tissue equivalent spherical proportional counter which is filled with tissue equivalent gases at low pressure (a few mm Hg). It has a sensitivity of greater than 20 rem/hr, and is used for absolute measurement of fast neutron absorbed dose in the energy range of 0.2 to 14 MeV. Interpretation of results obtained with these instruments involves extensive analysis, requiring the use of computer based multichannel analyzers.

4.6 SUMMARY

Proportional counters can be used to measure rest mass particles as well as photon radiation. They are especially useful for neutron measurements. If the quantity to be measured is the absorbed dose, proportional counters have tissue equivalent gas and wall composition, therefore, they can be used to obtain tissue absorbed dose directly. Gas flow proportional counters are most widely used for straight alpha and beta particle counting and fluence or fluence rate measurements.

Proportional counters are operated well within the gas multiplication region. The output pulse is proportional to the energy of the particle interacting with the fill gas. Gas multiplication factors from 10 to 10,000 are obtained.

As fill gases, noble gases either pure or in binary mixtures are used, with or without addition of polyatomic gases for stabilization and quenching. Argon is the most widely used noble gas, especially in the form of a 90% Ar - 10% CH_4 mixture (P-10 gas). Helium and heavier noble gases are used for special purpose applications. Also hydrocarbon gases, like methane, ethylene and propane are used as fill gases.

Survey instruments with large flat plate air or gas flow probes are used for measurement of alpha and beta contamination on surfaces. The windowless gas flow 2π system is most widely used as a laboratory counting system for detection of low and intermediate level alpha and beta activities. Systems for low energy x-ray and beta spectrometry are also available. Neutron detection and measurement systems utilize proportional counters extensively. Long counters are mainly used for secondary standards.

Calibration of proportional counters is usually performed with the goal of determining proportional counter efficiency, calibrating the response of wide range neutron survey meters, and generating neutron transfer standards. Many sources of all types are available for calibration purposes.

PROPORTIONAL COUNTERS

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CHAPTER 5

GEIGER - MUELLER COUNTERS

W. H. ELLIS

5.1 PRINCIPLES OF OPERATION

Geiger-Mueller counters, also known as Geiger or GM counters are a family of gas filled radiation detectors which are operated at voltages beyond the proportional counter region (see Figure 1.14). The principles of operation, general characteristics, configurations and applications of GM counters have been extensively treated quite early, along with ionization chambers and proportional counters, by Curran and Craggs (1949), Wilkinson (1950), and Sinclair (1956), and more recently by Price (1964), Sharp (1964), Emery (1966), Kiefer and Maushart (1972), and Knoll (1979). They have much larger gas amplification factors than proportional counters (10^9 vs 10^4). GM counters are by far the most numerous and commonly used instruments for radiation survey purposes. The popularity of the GM detector is due in part to its simplicity, ruggedness, low cost, high reliability and sensitivity. It is used extensively for survey and low-level counting in the detection and measurement of beta and gamma radioactivity and radiation fields. A single ion pair can initiate a fully developed output signal pulse in counters operated in the Geiger region. However, the resulting pulses are no longer proportional to the primary ionization, and all output pulses have approximately the same size. GM counters produce output pulses of the order of 1/4 to 1 volt compared to 1 to 10 millivolts for ion chamber operation. Therefore fewer complex electronic circuits are needed.

These relatively large pulses are capable of directly driving most rate meters and scalars without further amplification, although for various reasons this is not usually done. These characteristics not only account for the GM counter's wide popularity for high sensitivity applications but also its limitations with respect to exposure and absorbed dose rate measurement. In the former case, to be detected, the radiation needs only to produce a single ion pair within the sensitive volume of the chamber. The consequence is that for charged particles such as electrons, beta rays, protons, etc., these counters have effective counting efficiencies of close to 100% (for particles entering the sensitive volume). However, the chamber efficiency can be as low as 1% when the initial ionization is caused by secondary electrons produced by the interaction of gammas with the

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chamber walls or fill gas atoms. Even so, in this case, they are still at least two orders of magnitude more sensitive than ion chambers with comparable gas composition and chamber size. The output characteristics of GM tubes are fairly independent of the type of radiation and of its energy. The rates, shapes, and amplitudes of pulses generated are rather dependent on the geometrical characteristics of the counter (especially its length). This near independence of the type of initiating ionizing event in general prevents a relationship from being made directly to either absorbed dose or exposure. However, by adjustment of counter geometry, proper selection of wall materials, and/or the addition of filters, the device can be made to produce a response proportional to exposure (or kerma) over limited ranges of photon energy. Without these adjustments, counting rates are more or less proportional to fluence rate rather than to exposure (or kerma).

GM counters are most useful for measurements requiring high sensitivity and fast response characteristics such as rapid surveys and environmental monitoring for low-level contamination of working spaces. These features are also important when detecting "hot spots" or radiation leaks in shielding due to small cracks or pin holes. GM counters are also good for defining and mapping the extent or boundaries of radiation source fields in cases where knowledge of dose response characteristics is not necessary. However, their limitations must be recognized and caution exercised when knowledge of such characteristics is needed.

Another limitation characteristic of GM counters is that of "dead-time" counting loss (Curran and Craggs (1949), Sinclair (1956), Price (1964), Rollo (1977), ICRU 20 (1971)). These losses are incurred during an insensitive time period ("dead-time") which follows every counting event. This may range in duration from 10 to several hundred microseconds. During this time period the counter is unable to detect radiation even though this radiation has produced ionization within its sensitive volume. The dead-time is a temporary deactivation of the gas multiplication process. It is caused by a drop in the chamber collection potential due to a positive ion space charge left near the anode surface following the collection of electrons. The dead-time can be shortened by addition of certain gases or appropriate quenching circuits. The dead-time can be measured using a double-source method, therefore corrections for counting losses at high counting rates can be made. Nevertheless, GM counters should not be used when the counting rate is more than a few thousand counts per minute, which can be expected for radiation fields of about 50 to 100 mR/hr. As radiation levels increase, counter response first exhibits deviation from linearity and then at even higher levels count rate saturation occurs. Finally, as radiation levels rise still further, as shown in Figure 5.1, the counter becomes totally unresponsive, its output dropping back towards zero. Under such circumstances, careless use of GM counters can be hazardous since decreasing readings of counters which are in such an unresponsive (blocked) condition might be incorrectly

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interpreted as coming from a decreasing or even zero field. Use of smaller counter tubes and simple shielding techniques can minimize this problem by reducing the counter sensitivity; however, it cannot be eliminated entirely. Therefore great care should always be exercised for use in such conditions with backup ion chambers being used in conjunction with GM systems when there is a potential for encountering high levels of radiation. In general, a counter that gives no counts in several seconds should be presumed to be either inoperative or blocked. GM counters should not be used for the measurement of short high intensity radiation pulses. The dosimetry of pulsed radiation is described in the recently published ICRU 34, 1982.

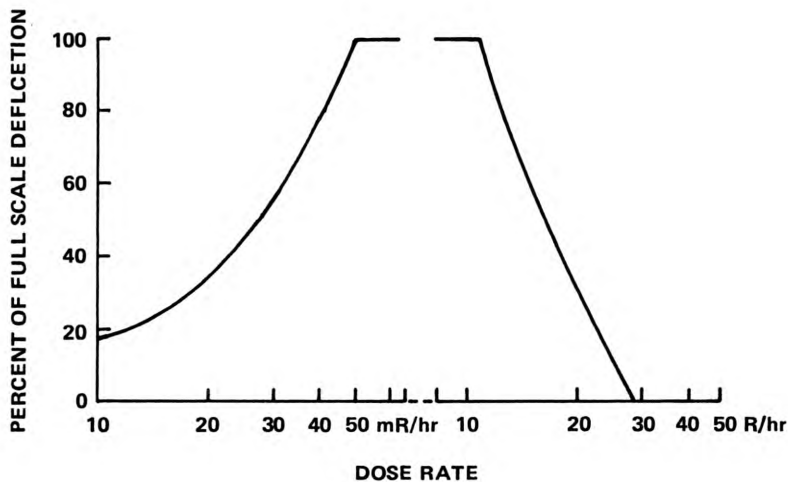


FIGURE 5.1 SATURATION OF A COUNTER-TYPE RADIATION MEASURING INSTRUMENT WITH INCREASING DOSE RATE.

Despite all of these limitations, GM counters still provide excellent indications of the presence of low level radiations. They yield results that are consistently reproducible within the limits set by counting statistics (ICRU 20, 1971). It should be pointed out here that although the output characteristics of the GM counter are independent of the type of incident radiation or energy, the probability that a given type of radiation will generate a response in the counter is energy dependent. In other words the sensitivity of the GM counter is energy dependent. However, the size and shape of the output pulse created by any ionization event is the same regardless of the amount of the initial ionization produced in the chamber. A GM counter may not respond at all to a 20 keV beta, but will give the same response to a 100 keV or 1 MeV beta. For this reason the counters should be calibrated with radiation with the same spectral composition as that of the radiation to be measured. This requirement may be relaxed if both the unknown and the radiation used for calibration have energies falling within the energy range for which the instrument is known to have an acceptably uniform response.

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Changes in sensitivity and energy dependence may occur from time to time due to aging, discriminator trigger level changes, etc., therefore, recalibration of GM counters should be undertaken periodically with a suitable standard source. If two standard sources having widely differing radiation energies are used and readings are consistent with those obtained with previous calibrations, it is unlikely that the spectral response has changed (ICRU 20, 1971).

5.1.1 Physical Construction

GM counter configurations can vary widely, depending on the particular application for which they were designed (Curran and Craggs (1949), Wilkinson (1950), Price (1964), Emery (1966), Eichholz and Poston (1979)). Nevertheless they consist of basically two main parts: the probe and the counter. A widely used arrangement for coupling these two components is shown in Figure 5.2. The anode of the probe (the GM tube) is capacitively coupled to the counter input through capacitor C_3 , and is connected to the high voltage power supply through a megaohm isolation resistor R_2 . The power supply provides the polarization voltage across the collector electrode (anode) and the cathode shell, while its d.c. component is blocked from the counter input by capacitor C_3 . This particular coupling circuit arrangement has the advantage of maintaining the cathode shell at ground potential, a condition which is particularly desirable to minimize electrical shock hazards when the metal envelope of the tube is the cathode. The current limiting resistor R_2 , in its series arrangement allows the voltage at the anode to drop temporarily and therefore the signal pulse to develop when electrons are collected and positive ions are drifting away from the anode following a discharge in the tube.

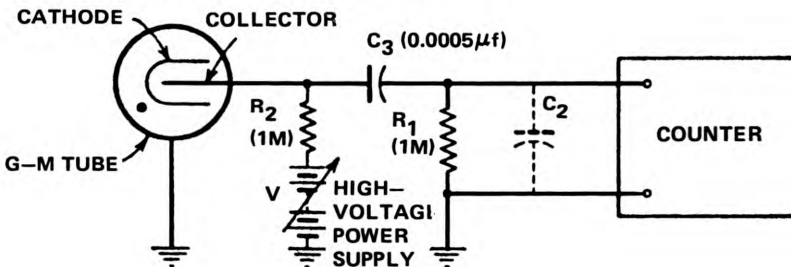


FIGURE 5.2 GM COUNTER INPUT NETWORK (PRICE, 1964).

Figure 5.3 is the equivalent circuit of Figure 5.2. The d.c. voltage component is eliminated from consideration and the parallel resistive and series capacitive components are combined into a single impedance network convenient for describing the signal pulse characteristics at the input of the counter. The signal voltage

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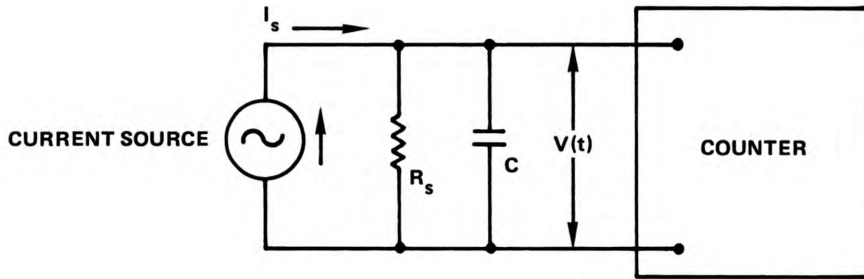


FIGURE 5.3 GM COUNTER EQUIVALENT CIRCUIT.

pulse profile (amplitude vs time) at the counter input will be determined by the values assumed by the lumped components of the circuit (identified here as R and C), and the time characteristics of the signal current pulse $I(t)$. The shape of the leading edge of the signal voltage developed across such a network is rather simple when the RC time constant is large compared to the current pulse width. It is the integral of the current divided by the capacitance of the network, i.e., $v(t) = (1/C) \int I(t) dt$. The fall-time characteristics of the resulting voltage pulse will be that of a slow exponential decay back to the base line ($v_{\max} e^{-t/RC}$). The shorter the RC time constants, the quicker the pulse will return to the baseline, decreasing the effective width of the pulse. These characteristics are exhibited in Figure 5.4.

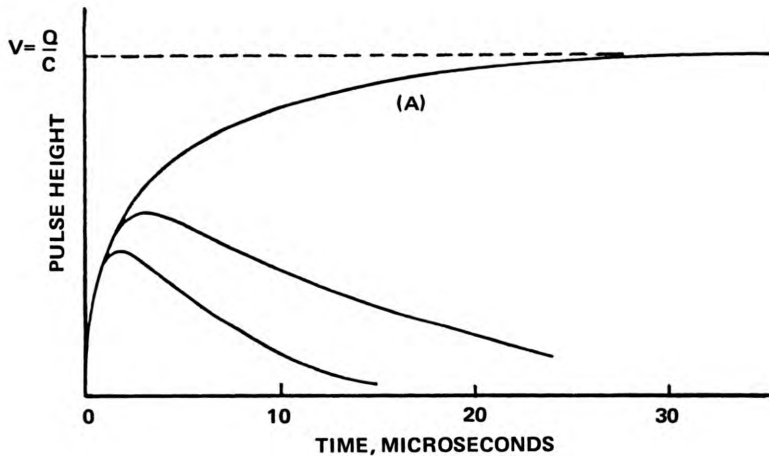


FIGURE 5.4 DEPENDENCE OF PULSE SHAPE ON TIME CONSTANT OF THE DETECTOR CIRCUIT. TOP CURVE (A) FOR $RC = \infty$; MIDDLE CURVE FOR $RC < \text{ION COLLECTION TIME } T$; BOTTOM CURVE $RC \ll T$ (EICHHOLZ AND POSTON, 1979).

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GM counting tubes are usually cylindrical, but other configurations are also quite common. Examples of typical configurations are shown in Figure 5.5. Their dimensions vary widely. Commercially available tubes range in lengths from 1 to 200 cm and diameters of 0.32 to 5.4 cm. The popularity and wide usage of coaxial electrode geometry within a cylindrical body configuration is associated with its relative ease of construction and the need for regions of high electric field strength for initiating the Townsend avalanche. However, other configuration are used for special purposes since the basic physical principle of the Townsend avalanche and GM discharge is independent of geometry.

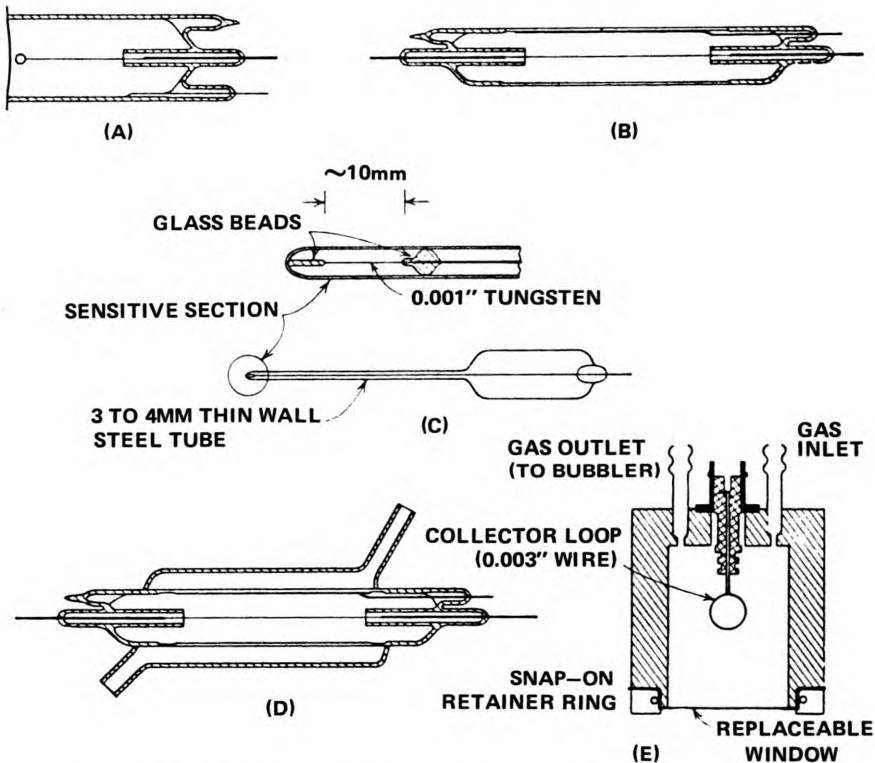


FIGURE 5.5 VARIOUS GEIGER-MUELLER TUBES. (A) BELL TYPE; (B) CYLINDRICAL TYPE; (C) NEEDLE COUNTER; (D) JACKETED TUBE; (E) FLOW-TYPE DETECTOR (PRICE, 1964).

There is considerable latitude in the choice of construction materials; however, modern detectors are usually made of either stainless steel or aluminum. The center wire (anode) is usually made of tungsten or stainless steel and is about 0.01 mm in diameter. Typically, the cathode is mounted coaxially with the anode as a part of the envelope of the tube. The wall thickness of portable beta-gamma survey meter tubes is typically 30 mg/cm². For low energy beta and some alpha capability, GM tubes have to be equipped with fragile 1 to 3 mg/cm² thick mica windows. A device more commonly

GEIGER-MUELLER COUNTERS

used in laboratory counting is the end-window counting tube shown in Figure 5.6.

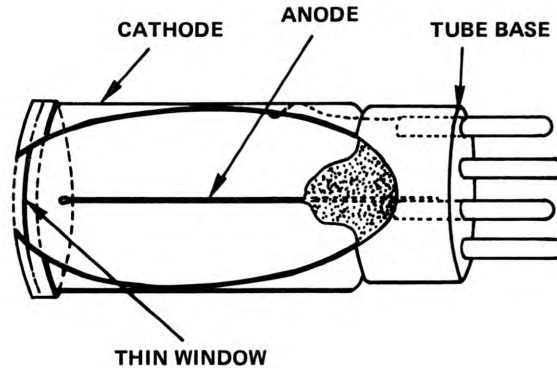


FIGURE 5.6 CUT-AWAY OF END-WINDOW GEIGER-MÜLLER COUNTER.

GM counting tubes are typically filled with a mixture of a rare gas and a trace quantity of a quenching agent with pressures ranging from 70 torr up to atmospheric (typically 90 torr He, Ar, or Ne). The quenching gases when used are either organic or halogen; their specific role in GM mode operation will be described later. The organic quenching agents generally exhibit more stable gas discharge performance. However, halogen quenching tubes have a much longer life expectancy, i.e., $> 10^{10}$ counts vs 10^8-9 counts, and are the chambers of choice for portable survey meters.

5.1.2 Geiger-Mueller Discharge Characteristics

GM mode operation is characterized by high sensitivity, uniformity of output, exceptional signal-to-noise performance, and counting rate stability. These capabilities, as well as its limitations are derived from the characteristics of the so-called Geiger discharge. Therefore, a brief review of this phenomena is in order at this point (Curran and Craggs (1949), Wilkinson (1950), Price (1964), Knoll (1979)).

When gas filled chambers are operated above the linear response range of the proportional mode the gas multiplication process caused by the Townsend avalanche still occurs. However, when the voltage is increased further the so-called Geiger region is reached. Here the Townsend avalanche initiates at much greater distance from the anode surface, greatly increasing the yield of the initial ionization avalanche. Finally, through ultraviolet photon ionization of fill-gas atoms or molecules, as illustrated in Figure 5.7, successive avalanches are generated, each being triggered by the preceding one. Consequently the discharge spreads into other regions of the chamber and so envelopes the full length of the anode surface with a highly ionized plasma-like column. Rapid collection of the highly mobile

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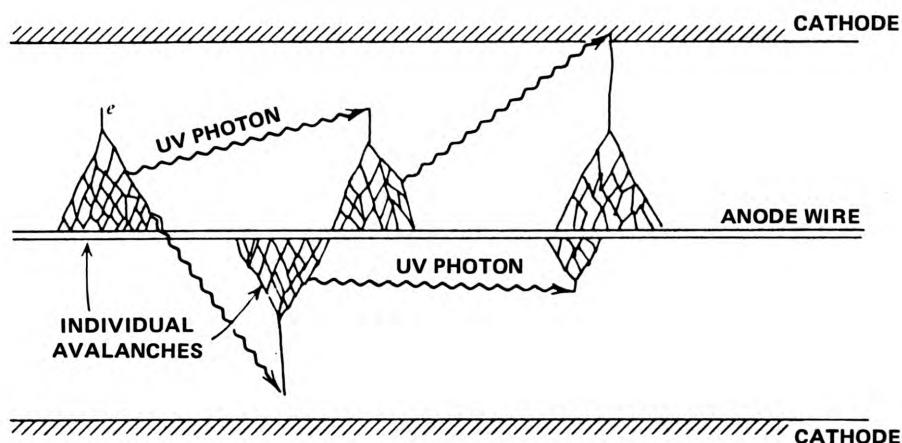


FIGURE 5.7 THE MECHANISM BY WHICH ADDITIONAL AVALANCHES ARE TRIGGERED IN A GEIGER DISCHARGE (KNOLL, 1979).

electron component of this plasma column leaves the slow moving positive ion space charge behind. This space charge lowers the electric field strength near the anode surface to below its critical value for ionization by collision. This terminates the initial avalanche discharge. Since the gas multiplication is stopped by the action of the space charge rather than by the fact that all charges are collected (as it is in the case of the proportional region) the Geiger discharge is (for a given applied voltage) always terminated after developing about the same total charge, regardless of the number of original ion pairs created by the incident radiation. Since the output signal is a direct function of this charge, all output pulses for a given applied voltage will therefore be about the same size and their amplitudes will convey no information about the properties of the incident radiation. However, halogen-quenching counters of small diameter are capable of passing currents of 100 microamperes or more. Consequently, specially designed small counters may be used with just a microammeter and a source of high voltage (approximately 500 volts) as an indicator of gamma radiation intensity. Such instruments are extremely simple and convenient to use where gamma-intensities up to many Roentgen per hour are likely to be encountered, such as when working with laboratory neutron sources or sources for gamma-radiography (Faires and Parks, 1973). A typical curve of current vs dose rate is shown in Figure 5.8.

Pulse Profile

When the discharge has been terminated, the positive ion sheath surrounding the anode, under the influence of the combined fields due to the applied voltage and that of its own space charge, drifts radially outward toward the cathode. As was discussed earlier for the proportional counter mode of operation, the outward drift of the positive ion sheath determines the major characteristics of the shape

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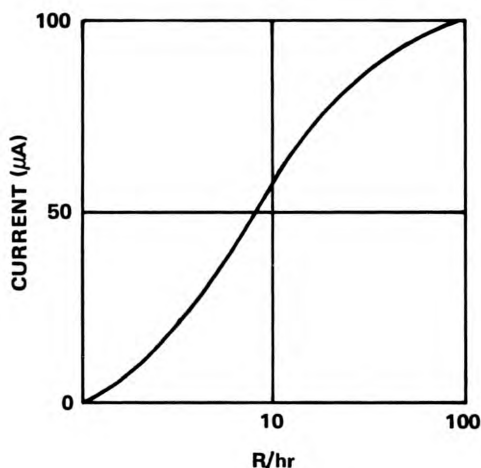


FIGURE 5.8 TYPICAL RESPONSE CURVE OF
HIGH-CURRENT GEIGER.

(or time profile) of the output signal pulse of the GM counting tube. Initially the pulse builds up very rapidly, as shown in Figure 5.4, as long as electrons are collected at the anode and the positive ions drift outward. However, most of its amplitude develops while the ions are still relatively close to the anode surface and within the intense field of this region. For a coaxial electrode configuration radial dependence of the field strength is $E(r) = v / \{r \ln(r_0/r_i)\}$, where r_0 is the radius of the outer and r_i of the inner electrodes. The ion drift velocity, v , is related to the electric field strength by $v = \mu E(r) / p$, where μ is the ion mobility and p is the gas pressure. The logarithmic dependency leads to an initial rapid drift and therefore to a sharply rising front edge for the observed pulse. During the later stages of drift of the ions to the cathode, the field strength decreases with radial displacement and so will the rate of rise of the signal pulse. The subsequent return of the pulse to the quiescent state is determined by the external circuit, as pointed out earlier (Figure 5.4). If the time constant of the collection circuit were infinite, it would make the full ion collection time on the order of 200–500 microseconds for the output pulse to reach its theoretical maximum. However, for optimum counting rate characteristics, time constants are most often chosen to be less than 100 microsecond. This largely eliminates the slow rising portion of the pulse and leaves only the leading edge. Even though a significant fraction of the output pulse amplitude may be lost, the signals are still sufficiently large for most applications.

Irrespective of the time constants used for the external circuitry, the counter is unable to produce another pulse following a previous one until the outward drift of the positive ion sheath has progressed sufficiently for the field in the vicinity of the anode to be restored to the Geiger discharge threshold region. During this

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time the tube is therefore "dead" and any additional ionizing events occurring will not be observed due to the absence of gas multiplication. Thus, radiation interactions that occur with the chamber or its walls during this so-called "dead-time" will be lost. A practical definition of the dead-time is the period between the initial pulse and the time at which a second Geiger discharge can be developed, regardless of its size. For most GM tubes this time will be of the order of 50-100 microseconds (Curran and Craggs (1949), Wilkinson (1950), Sinclair (1956), Price (1964), Fenyves and Haiman (1969), Rollo (1977), and Knoll (1979)). A pulse triggered immediately after the dead time will be quite small and may not actuate the recording system. However, with time, the amplitudes of these small pulses are seen to increase until at some point a full size pulse (discharge), as shown in Figure 5.9, will be observed to occur. The full recovery of the pulse indicates that the positive ions have been swept clear of the sensitive volume of the chamber and neutralized at the cathode surface. This latter time is known as the "recovery-time." The time characteristic of most practical interest is the "resolving-time," which is defined as the shortest time period between the initial pulse and the time at which a second Geiger discharge can produce a pulse that has an amplitude equal to or greater than the discriminator trigger threshold level. In practice the chamber dead-time and the counting system resolving time are often used interchangeably. However, this is not quite correct. It may lead to errors in estimating counter-system performance characteristics. The counting system resolving time must include not only the chamber dead-time, but must also account for the recovery-time characteristics which determine if the amplitudes have recovered sufficiently to exceed the discriminator trigger threshold. The dead-time is of the order of 50-200 microseconds, and is a characteristic of the counting tube, and cannot be affected by the operator. However, the resolving time is a function of both tube characteristics and discriminator trigger level setting. It can be

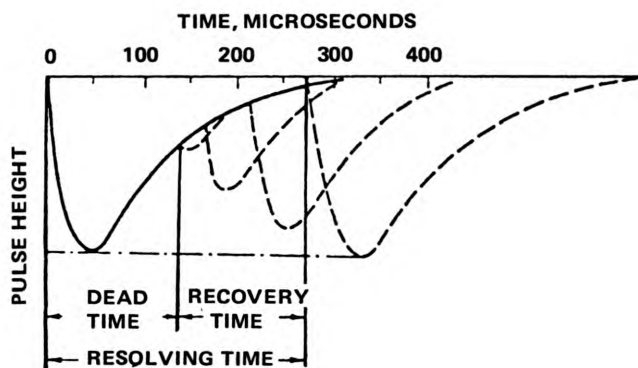


FIGURE 5.9 RELATION BETWEEN DEAD TIME, RECOVERY TIME AND RESOLVING TIME (EICHHOLZ AND POSTON, 1979).

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adjusted within the limits set by the system electronic noise and the maximum amplitude of the pulse.

Quenching

As positive ions produced during a Geiger discharge approach the cathode they are neutralized by electrons extracted from the metal cathode surface and are left in an excited state. There is a distinct probability that during their de-excitation process additional electrons will be promoted from the cathode surface and serve to retrigger the discharge. Whether or not this will occur depends on the nature of the fill-gas composition, the work function of the cathode material, the ionization potential of the ions, and the gas pressure. For high purity gases such as argon, where all of the ions reaching the cathode are of the same type, there are two mechanisms by which an additional electron can be emitted as a result of the neutralization process: 1) the energy difference between the ionization potential and the work function of the cathode may be radiated away as a photon which in turn may create a photoelectron, 2) the energy difference may be used directly in a radiationless liberation of an electron from the cathode. This additional electron will result in another discharge, with the possibility of continuous recycling, unless provisions are made to prevent it, or "quench" it.

Multiple discharges may be prevented by circuits external to the GM chamber or by use of self-quenching chambers (Curran and Craggs (1949), Wilkinson (1950), Sinclair (1956), Fenyves and Haiman (1969)). In the former case the circuits are designed to drop the voltage across the chamber for a period sufficiently long to allow the ions to be collected at the cathode and sufficiently below the value needed to maintain the discharge. The effect is that the discharge is not retriggered during the collection time. One method to accomplish this is to use a large series resistor (typically 10^8 ohms) to provide a millisecond time constant for the collection circuit. However, the disadvantage is that several milliseconds are required for the anode to return to its normal working voltage. Thus, full Geiger discharges for each event are produced only at very low counting rates.

Another method is to use the multivibrator type external quenching circuits, which drop the anode voltage below the Geiger threshold. They have been used with non-quenching or with self-quenching tubes in order to shorten the dead-time. Values as small as 40 microseconds have been reported. Circuits which temporarily reverse the polarity of the operating voltage at the anode have reduced the dead-time to as low as 20 microseconds, which is ten times smaller than that normally found with self-quenching tubes.

It is common practice to prevent multiple pulsing through the addition of a second component (or quenching agent) to the fill gas (Curran and Craggs (1949), Wilkinson (1950), Sinclair (1956), Fowler et al. (1959), Price (1964), Sharpe (1964), Emery (1966), Fenyves and Haiman (1969), Hendee (1970), Rollo (1977), and Knoll (1979)). The

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quench gas usually has a relative concentration of 5–10 percent, has a lower ionization potential, and is a more complex molecule than the gas of the primary component. Two different types of self-quenching tubes are available, namely those using a halogen gas and those which use organic compounds as quench gas. In these GM tubes there is a transfer of charge from the positive ions of the primary gas to the molecules of the quench gas. When these charged molecules reach the cathode they extract electrons from the surface of the cathode, but this process results in the dissociation of the quench gas molecules rather than the production of additional electrons at the cathode surface.

Typical organic quench gases are polyatomic gases, including alcohol, ethyl formate, and amyl acetate. A gas-fill mixture frequently used is argon at a pressure of about 90 mm Hg (12 kPa) with ethyl alcohol vapor, or similar organic gas, at a pressure of about 10 mm Hg (1.3 kPa). Alcohol has a lower ionization potential than argon; also the charge transfer is very efficient. Therefore, after a few collisions, the charge will consist of only alcohol ions. In contrast to argon ions, which after recombination into excited atoms lose their excess excitation energy by photon emission or collisional de-excitation (as depicted in Figure 5.10) alcohol ions dissociate and do not produce photoelectrons at the cathode. For the organic quench gases the dissociation is irreversible and therefore the life of an organically quenched GM counter is limited. The useful lifetime is typically 10^8 to 10^{10} pulses; beyond this point multiple pulsing can no longer be prevented due to quench gas depletion.

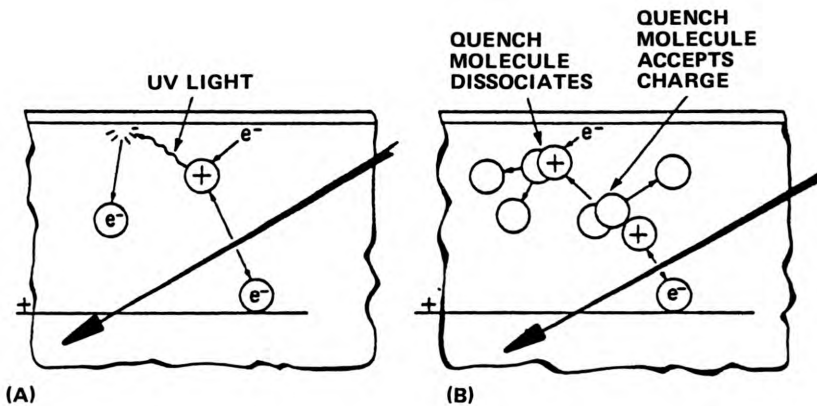


FIGURE 5.10 OPERATION OF GEIGER-MUELLER COUNTER: (A) FORMATION OF INITIAL ION PAIR; (B) EFFECT OF QUENCH MOLECULES.

Bromine and chlorine gases are commonly used in halogen-quenched GM counters. In this type of counter the quench gas molecules reform after some time and therefore these counters have a long lifetime, determined by leakage of the tube seals.

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5.1.3 Geiger-Mueller Counter Electronics

Since the GM discharge provides an output ranging from 0.25 to 10 volts (and in some cases as large as 100 volts), the counting circuit electronics can be relatively simple for both laboratory counting and portable survey instruments. In principle neither of the two types of counters requires preamplification of the input signal, although in some special applications preamplifiers are used nevertheless.

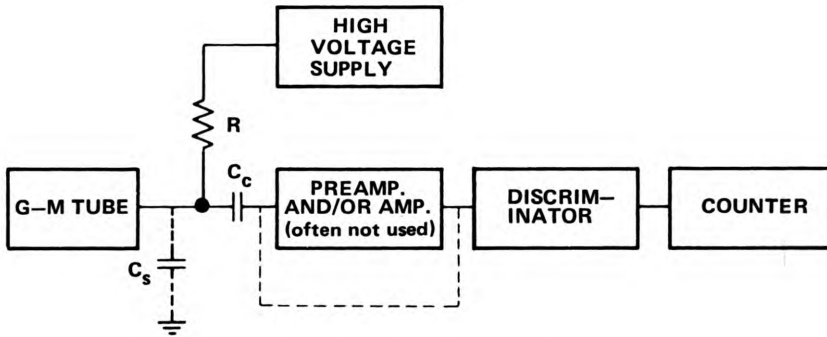


FIGURE 5.11 BLOCK DIAGRAM OF THE COUNTING ELECTRONICS NORMALLY ASSOCIATED WITH A GM TUBE. THESE FUNCTIONS ARE OFTEN COMBINED ON A SINGLE CHASSIS TRADITIONALLY CALLED A "SCALER" (KNOLL, 1979).

Figure 5.11 shows a block diagram of a typical circuit used for laboratory counting. The system is comprised of a discriminator and scaler in addition to the coupling circuit components described previously. Many times a discriminator is provided as a component of the scaler or ratemeter. However its trigger level adjustment is not always accessible to the operator. The main difference between the circuitry for laboratory counting and that for survey meters is the use of pulse rate (analog) meters in place of (digital) counters. The latter requires a trigger pulse shaping circuit with an adjustable output pulse width control. In general the (analog) meters read cpm (counts per minute) and/or mr/hr, while the (digital) counters read just counts. Of course such a counter can be set with a timer to read for one minute if cpm information is desired.

The example shown in Figure 5.12 is typical of the coupling arrangement used with portable survey meters, while Figure 5.13 is a photograph of the same system. A probe which is visible on top of the case contains the GM tube within a protective shield. It is provided with a mechanical shutter that can be opened for increased beta ray sensitivity, or closed for detection of gamma rays only. Halogen quenched chambers have a high value resistor (2 to 20 megaohms) R_s , in series with the anode to minimize the effect of the capacitive loading of the connecting cable and other parasitic capacitances. Such a resistor (in effect) extends the counting plateau. Without this resistor halogen counters are non self-quenching at higher voltages.

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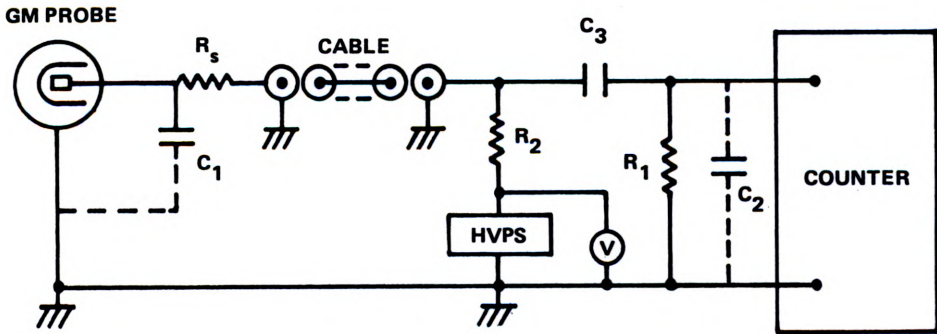


FIGURE 5.12 GM SURVEY COUNTER INPUT CIRCUIT.

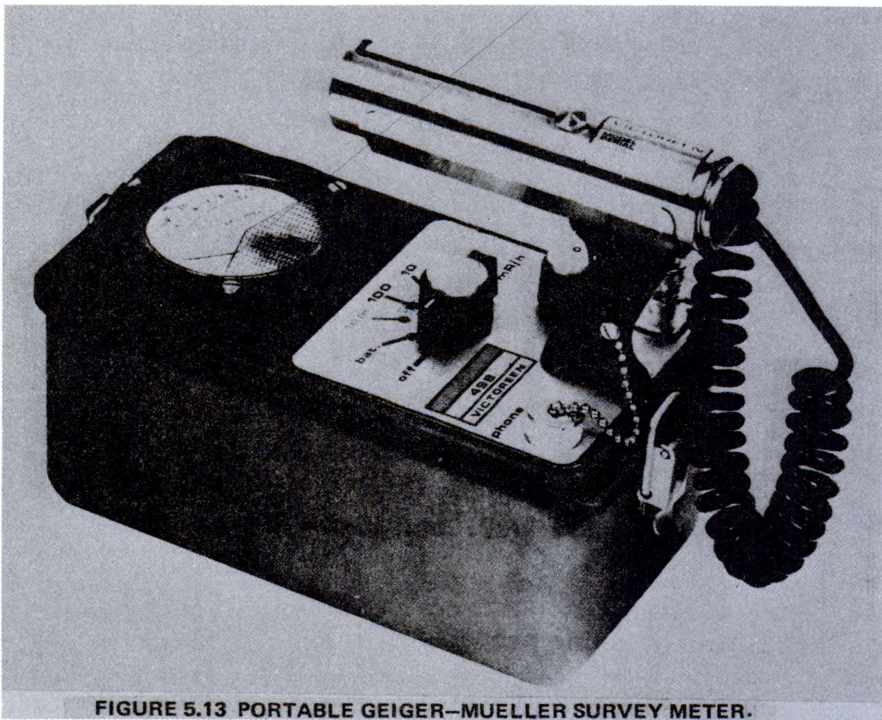


FIGURE 5.13 PORTABLE GEIGER-MUELLER SURVEY METER.

In general the "counter" section of the circuitry is contained within a small case or box, which also contains the battery operated high voltage power supply. The front or top panel of the counter case has a visual indicator, such as a panel meter, or digital readout for display of exposure or count rate. Output pulse coupling to earphones or a speaker to provide an audible indication of count-rate response is usually provided. Such audible systems serve as highly sensitive and rapid means for alerting the operator of changes in the radiation field sensed by the GM probe. Probe and counter are connected by a cable that serves the dual purpose of high voltage lead and signal cable. Portable survey meters usually have batteries

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which can be recharged. They provide the energy for the electronic high-voltage power supply. For this type of electronics, warm-up time is negligible. Laboratory systems are usually powered by line voltage. For most analog readout systems response-time-to-full-scale ranges from 0.8 sec for low exposure scales to 45 seconds for extremely high scales, 1 to 2 seconds being typical.

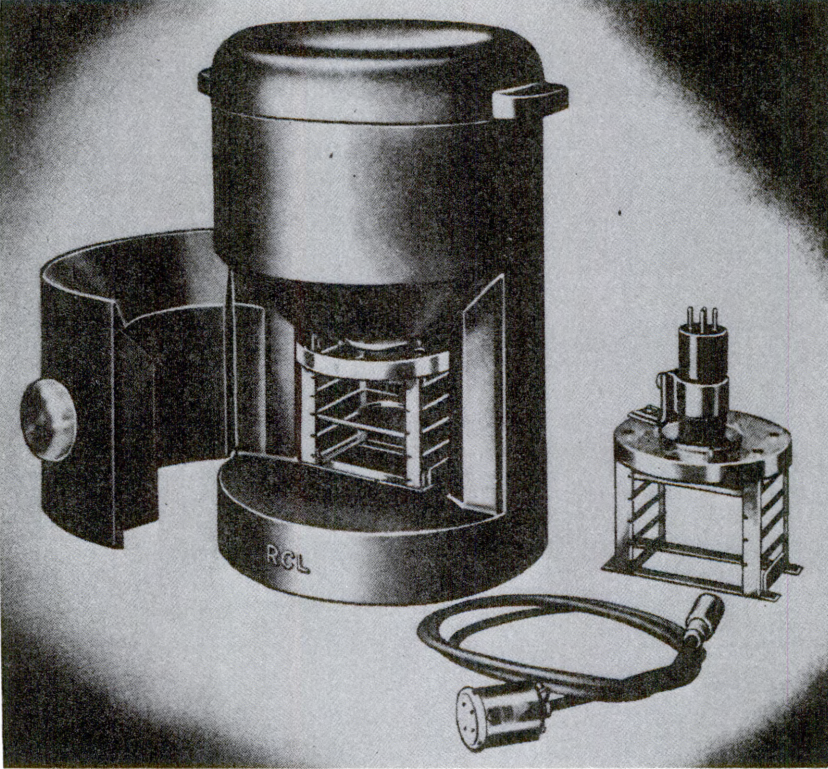


FIGURE 5.14 TYPICAL LABORATORY SYSTEM FOR SOURCE-DETECTOR COUNTING.

Variations in the configurations described above are numerous but all GM counters have similar components. Examples of common laboratory counting arrangements used for the measurement of beta and gamma activities are shown in Figures 5.14 and 5.15. The former figure exhibits source-detector counting and the tube holder-source mount arrangements, while the latter shows a lead enclosure containing a thin end-window GM tube mounted in a sample holder with shelves at various distances from the counter window.

5.1.4 Counting Characteristics

The performance capabilities, such as gas amplification and proportional gain saturation of GM tubes are best reflected in response curves relating output signal pulse height to applied voltage, as illustrated in Figure 5.16, while the overall system

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performance is best demonstrated by a plot of counting rate response to a radiation source as a function of applied voltage (see Figure 5.17). Figure 5.16 shows the general nature of the former characteristic for a helium-alcohol filled, bell-type, thin window GM tube which is widely used for laboratory type measurements of alpha and beta radioactivities. The figure illustrates the pulse size dependence of gas filled chambers on applied voltage for different degrees of initial ionization, represented here by alpha and beta particle responses. As the voltage across the chamber is increased the chamber response progresses from the ion chamber saturation region, through the proportional counter region, and finally into the GM counter saturation region. An appropriate discriminator trigger threshold level setting is indicated by the dashed line. Figure 5.17 shows a GM threshold below which no counts are observed. Above the threshold the counting rate rises rapidly

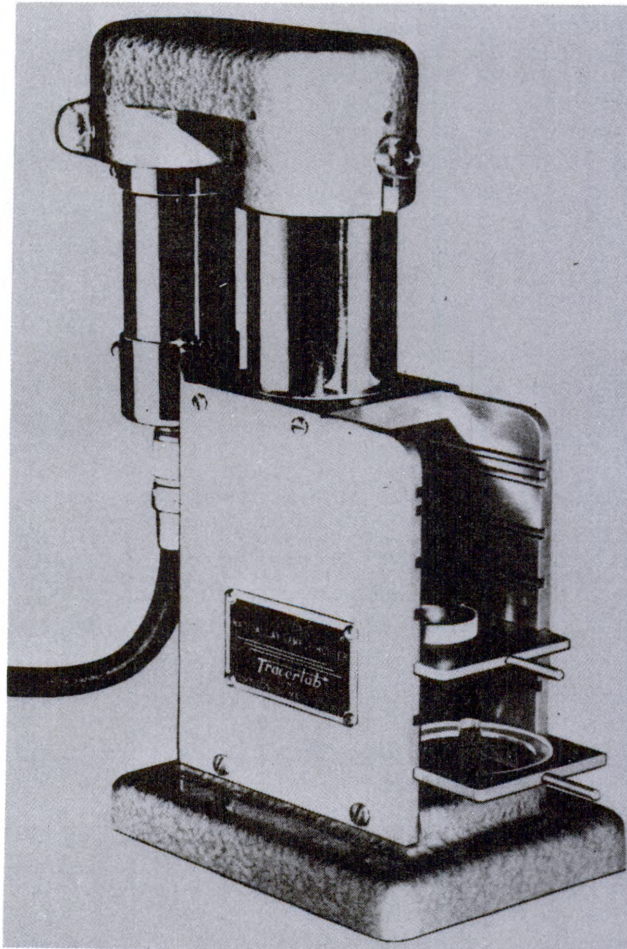


FIGURE 5.15 GEIGER-MUELLER DETECTOR WITH ADJUSTABLE
SAMPLE HOLDER MOUNTED IN A LEAD ENCLOSURE.

GEIGER-MUELLER COUNTERS

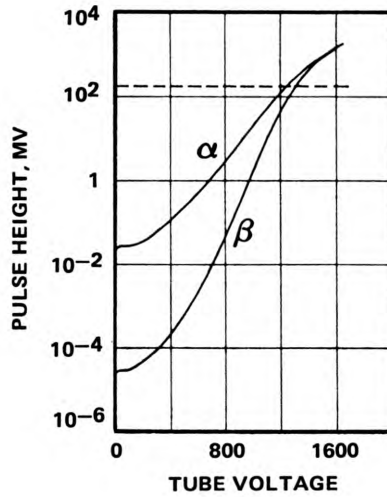


FIGURE 5.16 PULSE HEIGHT VERSUS TUBE VOLTAGE FOR A BELL-TYPE GM TUBE FOR BOTH ALPHA AND BETA PARTICLES; DASHED LINE INDICATES DISCRIMINATOR SETTING (PRICE, 1964).

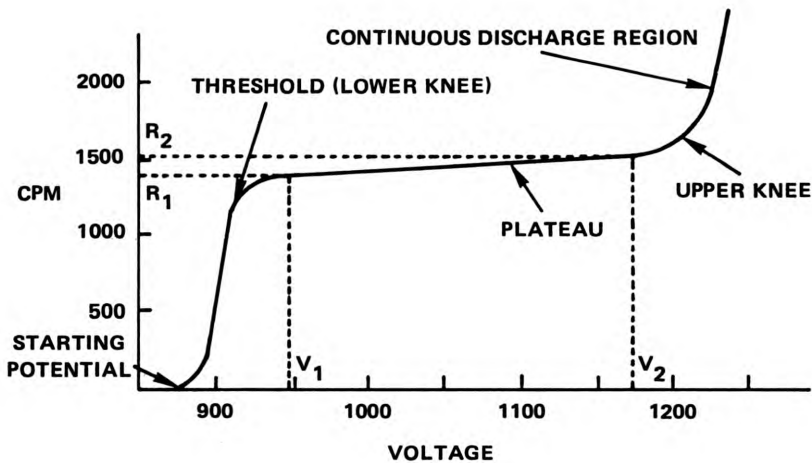


FIGURE 5.17 CHARACTERISTIC CURVE FOR GEIGER-MÜLLER TUBE.

until a plateau is reached (V_1) where the system counting rate remains relatively constant for several hundreds of volts, and then at V_2 it increases rapidly until the applied voltage exceeds the continuous breakdown discharge value (not shown in the figure). In order to conserve the lifetime of the tube, the chamber should be operated at the low voltage side of the plateau. As can be seen in Figure 5.16 there is a slight difference in the alpha and beta threshold voltages. This arises from locating the discriminator trigger level slightly below the GM saturation point. The alpha particle's greater initial ionization yields a slightly larger pulse

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for a given applied voltage, causing the alpha particle pulses to exceed the discriminator trigger level at lower voltages than occurs for beta pulses. The plateau always has a slight slope, which is due in part to an extension of the sensitive volume of the chamber as the voltage increases. Also contributing to the slope are occasional failures of the quenching action, which are more likely to occur at higher voltages. Slopes of less than 5 percent per 100 volts are desirable, although up to 10 percent per 100 volts are usable in many applications. Organic quenched tubes tend to show the flattest plateau, with slopes of the order of 2-3 percent per 100 volts change in the applied voltage. The applied voltage is usually lower for halogen quenched tubes than it is for organic quenched tubes. For example, argon-alcohol counters usually require a starting voltage of at least 1000 volts, but neon filled tubes quenched with bromine have starting voltages as low as 200-300 volts.

As noted earlier, the counting efficiency of GM counters for charged particles entering the sensitive volume of the chamber approaches 100 percent. In most practical situations, the effective counting efficiency is therefore determined by the probability that the incident radiation penetrates the window of the tube without absorption or backscattering. With alpha particles, absorption within the window is the major concern, therefore windows with thickness as small as 1.5 mg/cm^2 are commercially available. Generally, thicker windows can be used for counting beta particles, but some will be backscattered from the window without penetrating if the window thickness is a significant fraction of the electron range in the window material.

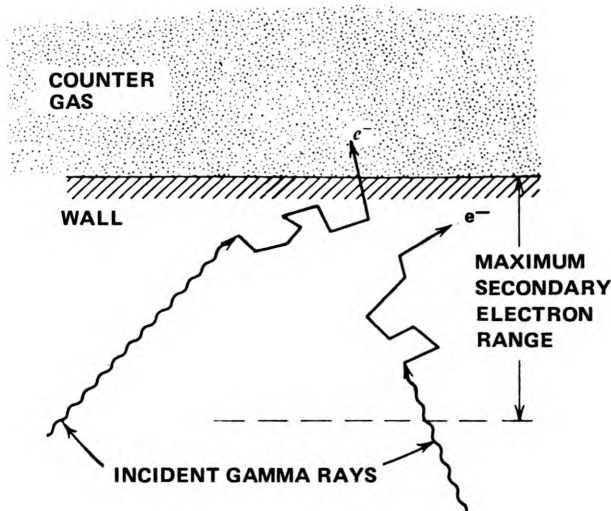


FIGURE 5.18 GAMMA-RAY INTERACTIONS RESULT IN CREATION OF SECONDARY ELECTRONS IN COUNTER WALL. OUTPUT PULSES RESULT ONLY WHEN INTERACTION OCCURS WITHIN AN ELECTRON RANGE OF WALL (KNOLL, 1979).

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In the case of gamma rays, the GM counter response comes about by way of gamma ray interaction in the solid wall of the counter (Sinclair (1956), Emery (1966), Kiefer and Maushart (1972)). If the interaction takes place close enough to the inner surface of the wall so the secondary electrons which are created in the interaction can leave the surface with sufficient energy to ionize the gas and create electrons and ions, a pulse will result. Because only a single ion pair is required, the secondary electron need only barely emerge from the wall surface, near the end of its track, in order to generate a pulse in the Geiger tube.

Counting efficiency for gamma rays therefore depends on two separate factors: 1) the probability that the incident gamma ray interacts in the wall and produces a secondary electron, and 2) the probability that the secondary electron reaches the fill gas before the end of its track in the solid, as indicated in Figure 5.18. This happens only for a region of typically a millimeter or two. Making the wall thicker than this value contributes nothing to the efficiency since electrons formed in regions of the wall farther from the gas have no chance to reach the gas and initiate a discharge.

The probability of gamma ray interaction within this critical layer increases with the atomic number of the wall material. For this reason GM tubes with bismuth ($Z=83$) cathodes have been widely used in the past for gamma-ray detection. The probability of interaction within the sensitive layer remains small even for high- Z materials, and typical gamma ray counting efficiencies are seldom higher than several percent.

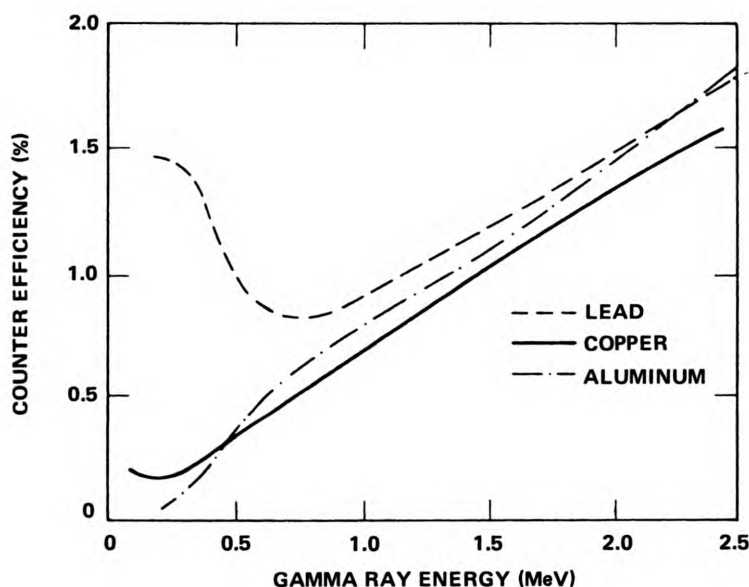


FIGURE 5.19 THE EFFICIENCY OF GM TUBES FOR GAMMA RAYS NORMALLY INCIDENT TO THE CATHODE (SINCLAIR, 1956).

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Figure 5.19 shows the dependence of the efficiency on gamma photon energy. The almost linear rise at high energy reflects the fact that the interaction cross section (principally due to Compton scattering) changes slowly, while the range of the secondary electrons increases almost linearly with photon energy. This is due to the fact that electrons can now reach the sensitive gas volume from greater depths within the cathode wall. The difference in efficiency for various cathode materials for low energies is also due to difference in Z . The rise at low energies is due to the rapid increase in the number of photoelectrons and is more pronounced for high Z cathode materials. However, the increased photoelectron production will also lead to attenuation of the incident beam. (This explains the decrease in the efficiency vs energy relationship for a lead as shown in Figure 5.14.)

At low photon energies, where the photo-electric effect may be appreciable, direct gamma ray interaction in the fill gas of the tube may no longer be negligible. Therefore, the counting efficiency for low energy gamma rays and x rays can be enhanced by using gases of high atomic number at higher pressures. Xenon and krypton are common choices for these applications, and counting efficiencies near 100 percent can be achieved for photon energies below about 10 keV.

A more useful method for showing the dependence of the sensitivity on photon energy is to present the sensitivity of a counter as a function of exposure, or kerma. In principle this can be obtained by dividing the counter efficiency by the photon energy which yields counts per unit energy fluence. From this, the sensitivity in terms of exposure and other dosimetric quantities can be obtained. Figure 5.20 is an example of such a characteristic. It shows the counting efficiency per cm^2 normalized to an exposure of 1R, as a function of gamma photon energy. There are energy regions where this characteristic is relatively flat for a given material and does not change drastically from material to material, but there are other energy regions where substantial differences are exhibited in the characteristic. They are mainly due to differences in the degree of importance that photoelectric and Compton scattering effects play in the role of determining the total gamma ray interaction cross sections of these materials for that particular photon energy. Knowledge of these types of characteristics for materials allow the design of chambers which have essentially flat energy dependent exposure responses from approximately 300 keV to 3 MeV. This is accomplished by use of composite materials and special geometrical configurations. For general practice, however, the implications are that GM counter instruments can be used for exposure measurements for gamma ray sources whose energies fall within the "flat" response range. Measurements involving sources with unknown energy characteristics or with characteristics which fall outside the flat calibration range have to be scrutinized carefully since the results obtained can be in considerable error and should be used only as an indication of the presence of radiation fields or as a crude approximation with respect to exposure rate.

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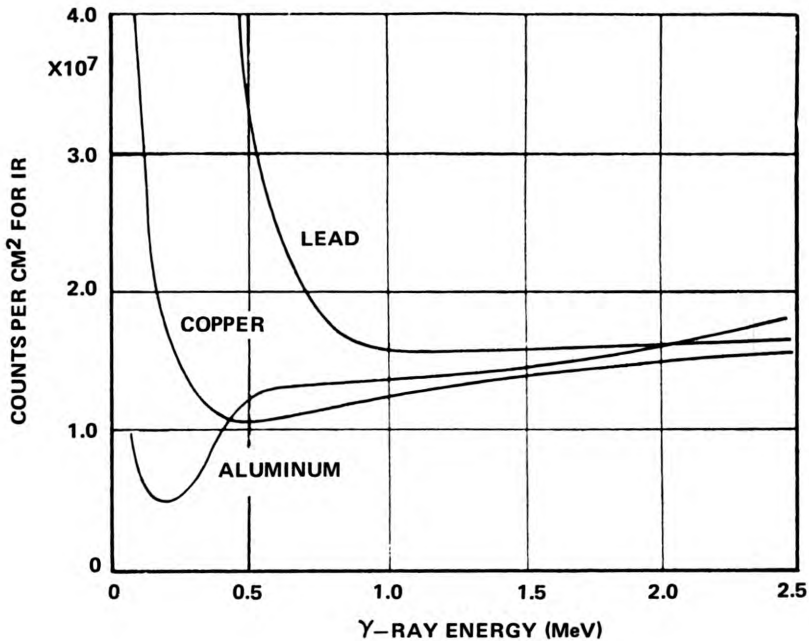


FIGURE 5.20 RESPONSE OF GM COUNTERS OF DIFFERENT CATHODE MATERIALS PER ROENTGEN OF GAMMA-RAY EXPOSURE (SINCLAIR, 1950).

Chrome-steel, a widely used cathode material for commercial GM tubes, exhibits characteristics close to those shown for copper in Figure 5.19 and 5.20. A marked rise in the sensitivity due to increased efficiency of the photoelectric effect is observed at low energies. This characteristic can be seen clearly in Figure 5.21 in the case of the unshielded probe. The modification and reduction of the peak shown to occur at about 50 keV for the probe with the shield closed was brought about by use of a cover of additional shielding material. These characteristics are further shown in Figure 5.22 for the shielded and unshielded energy dependent sensitivity response of two commercial GM survey meters. Commercial GM tubes of this type generally exhibit reduced sensitivity in the direction of the counter axis, as shown in Figure 5.23. Graded filters (or shields) consisting of materials such as lead or tin have been used to flatten out the response curve and bring the end response of such probes to within 90% of the side response. The directional response characteristics of such probes make it essential that the probe source arrangement used for calibration be duplicated as closely as possible for the actual measurement if accurate results are required.

Perforated or screen type filters (or shields) have been used successfully in superimposing varying portions of the screened and unscreened response curves. Probes with removable shields for increasing the low energy gamma and x-ray photon sensitivity also have increased beta sensitivity making it possible for tubes to have common beta-gamma utilization.

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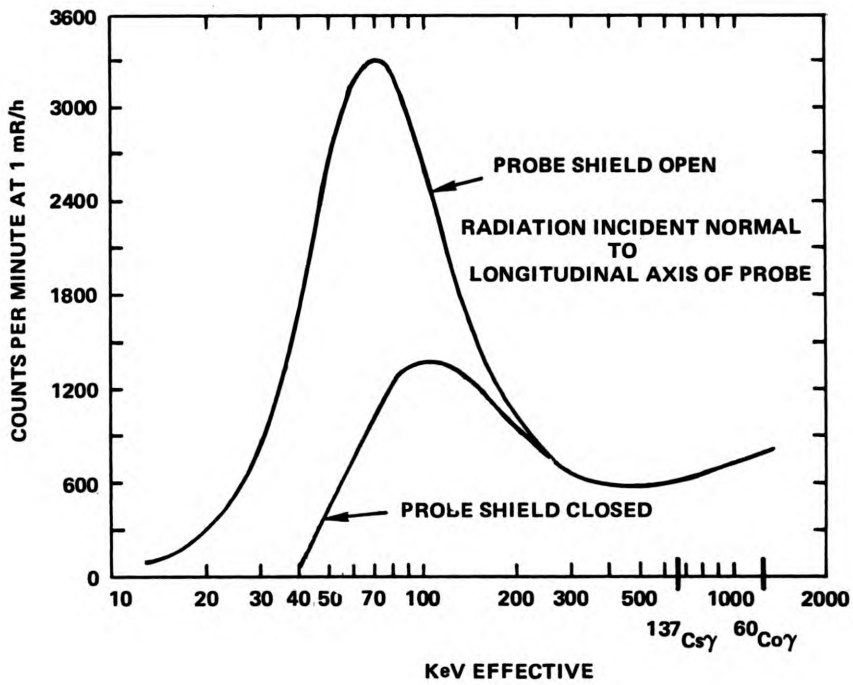


FIGURE 5.21 GM MODEL 493-50 PROBE WITH MODEL 493 COUNTER.

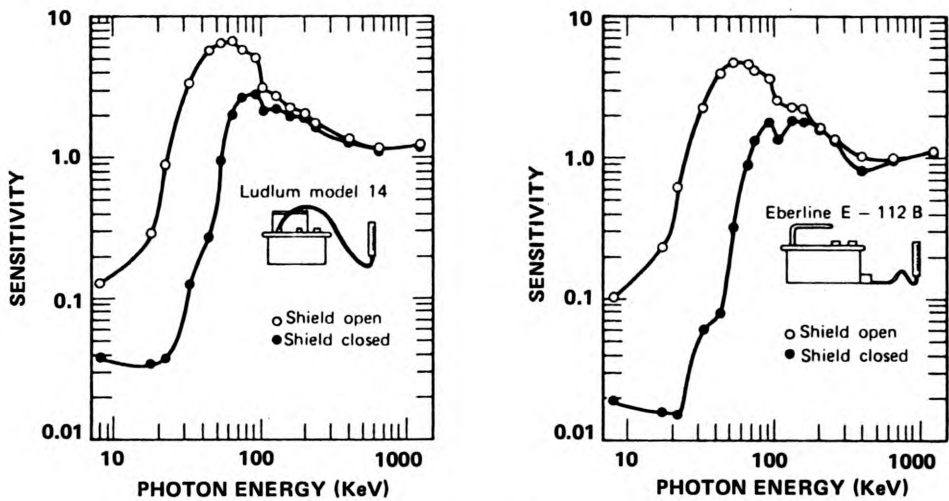


FIGURE 5.22 THE SENSITIVITY VERSUS ENERGY OF TWO COMMERCIALY MANUFACTURED GEIGER SURVEY METERS. SENSITIVITY IS DEFINED AS THE INDICATED EXPOSURE RATE DIVIDED BY THE TRUE EXPOSURE RATE (STORM ET AL., 1974).

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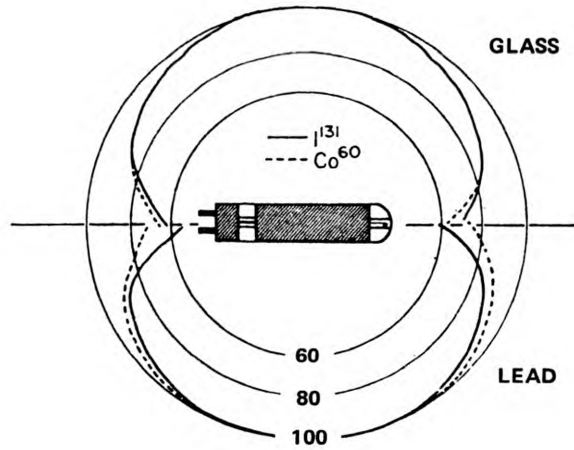


FIGURE 5.23 DIRECTIONAL SENSITIVITY OF GAMMA-RAY COUNTERS WITHOUT ADDITIONAL SCREENING (EMERY, 1966).

Problems associated with the directional and energy dependence of end-window probes are further illustrated in Figure 5.24. The thin window allows for strongly peaked low energy sensitivity below 200 keV while the thicker side-wall attenuates the low energy component and yields a relatively flat response down to approximately 80 keV with a sharp drop off thereafter. An appropriate cover for the end window of the probe should partially remove the directional asymmetry characteristics. However, care must still be exercised both in calibration and use of such systems.

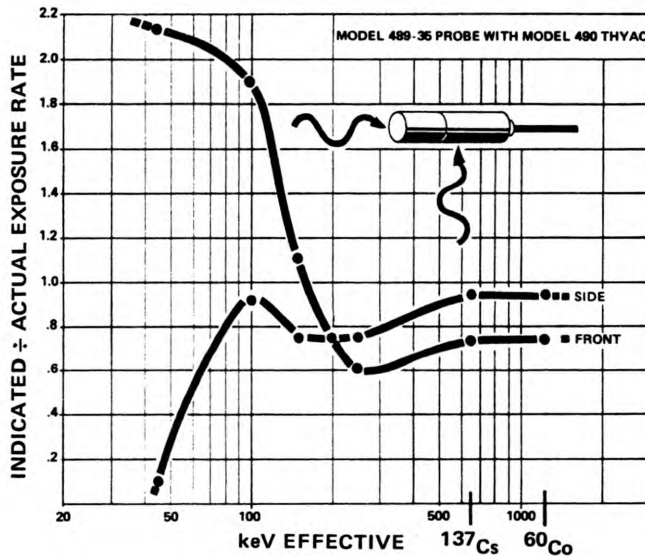


FIGURE 5.24 GEOMETRICAL EFFECTS IN AN END-WINDOW GM COUNTER.

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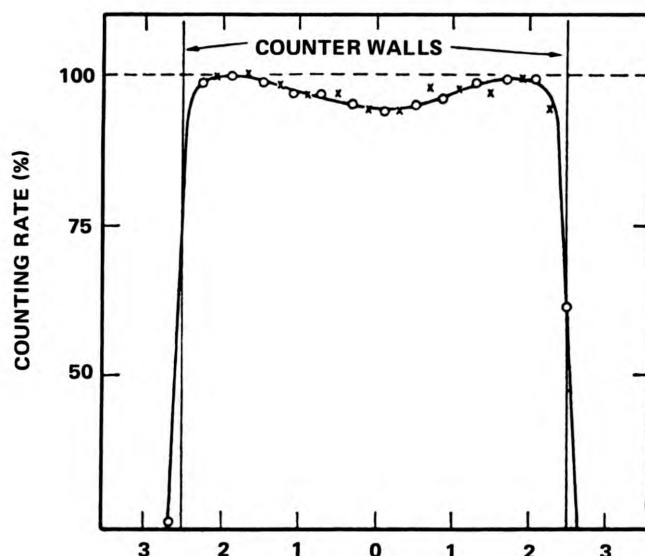


FIGURE 5.25 VARIATION IN COUNTING RATE AS A COLLIMATED BETA-PARTICLE SOURCE IS MOVED ACROSS A DIAMETER OF A LARGE END-WINDOW COUNTER (SINCLAIR, 1956).

Directional characteristics are also experienced with the thin end-window probes used for alpha-beta and low energy gamma and x-ray counting (see Figure 5.25). They are also caused by the large difference between the thicknesses of the end-window and the side-wall. These directional properties are modeled by assuming that the area effective for collection is the projection of the window in the direction of the incident radiation. The result of such a modeling procedure is that the end-window counter obeys a cosine law. The empirical response characteristics are illustrated in Figure 5.25 for a collimated beta ray source which is moved across the diameter of the end-window of the counter tube (see Figure 5.26). The drop in sensitivity exhibited for the central region of the window is caused by the bead at the end of the central electrode wire. Figure 5.27 illustrates similar characteristics for the side wall response for a collimated gamma-ray source.

5.2 GM COUNTER CAPABILITIES AND NORMAL USE

GM counters are used for monitoring and counting low and high levels of radiation. They have long life, and when used within their limitations provide adequate instrumentation for a variety of tasks. However, they should not be used for accurate measurement of dose.

GM counters are sometimes used to estimate exposure rates where the rate is low enough for the resulting inaccuracy of its measurement to be unimportant, or when the photon energy is the same as the one used for calibration. Specifications for construction,

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test, and calibration of portable GM counters are found in ANSI N13.4-1975.

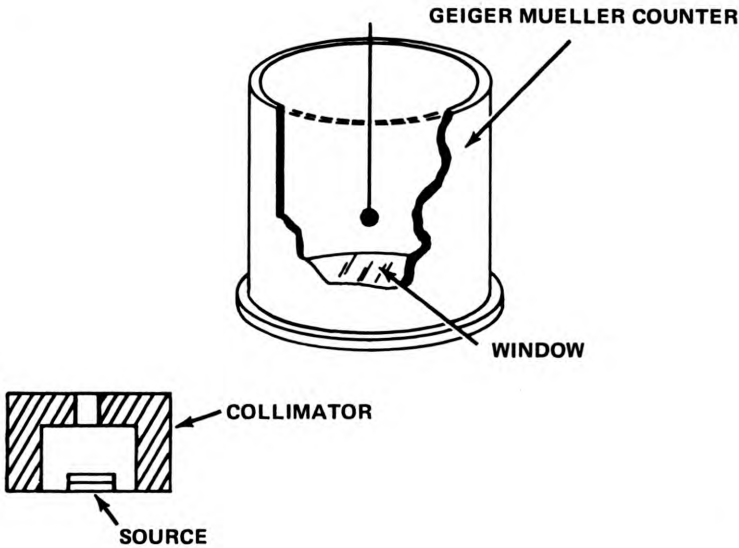


FIGURE 5.26 TEST SETUP FOR RADIAL SENSITIVITY DETERMINATION.

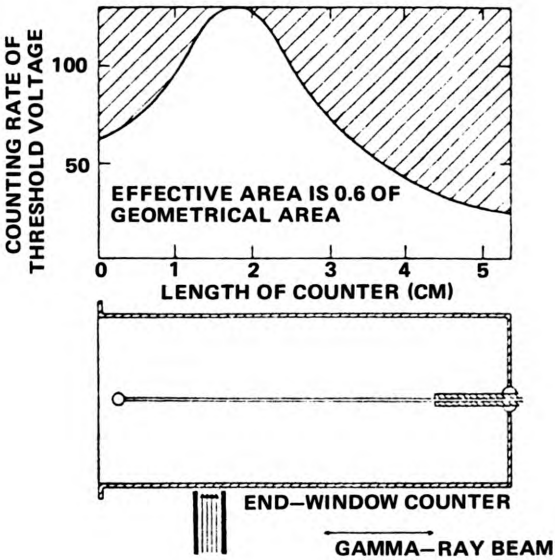


FIGURE 5.27 RELATIVE COUNTING RATE FOR COLLIMATED GAMMA-RAY SOURCE MOVED PARALLEL TO THE AXIS OF AN END-WINDOW GM COUNTER (SINCLAIR, 1956).

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Survey meters for the detection of the presence and measurement of the approximate level of high energy gamma ray exposure are available in ranges as low as 0.01 mR/hr to as high as 1000 mR/hr (Sinclair (1956), Fowler et al. (1959), NCRP 57 (1978)). The low exposure rate instruments are highly energy dependent while the high exposure rate devices are shielded so that their response is relatively flat (20%) over a useful energy range above 40 keV. However, even low level GM counters can observe gamma ray source energy down to 6 keV only in an inefficient manner.

5.2.1 Survey Meters

X- and gamma-radiation field survey meters can use GM tubes to count photons whose secondary electrons reach the counter gas to produce at least one electron-ion pair. GM counters are the detectors usually employed in portable radiation field survey and monitoring equipment where accuracy is not required. Their principle advantages are high sensitivity, small size, and simple reliable electronic circuitry (Kiefer et al. (1969), ANSI N13.4 (1975), NCRP 57 (1978)).

Beta and alpha particle "pancake" or thin window detectors are available for the counting of beta particles above 35 keV and alpha particles above 3.5 MeV. "Moveable shield" beta particle window probes have capabilities of counting beta particles with energies above 200 keV.

Personal "dosimeters" using miniature GM tubes are available that indicate in digital display and audible chirps the exposure (50 keV to 1 MeV). These devices also have integrated exposure capabilities and saturation alarms. The whole system can be packaged in a shirt pocket clip case.

Survey meters should not be used to monitor x-ray scatter because their energy nonlinearly provides erroneous measurement of low energy gamma exposure.

5.2.2 Counters and Fixed Location Contamination Monitors

Due to the high sensitivity of GM counters they are used to indicate the presence of low level contamination and perhaps sound an alarm. Typical instruments include: hand-operated thin window "frisking" detectors, and hand, foot, and portal monitors. Each of the detectors is sensitive to gamma rays, beta and alpha particles. GM counters are also used for laboratory assay work. In this technique a thin window GM tube is placed inside a shield and the sample to be assayed (or counted) is also placed inside the shield. With a known geometry, an accurate assay can be performed.

GM counters are used as area monitors to indicate exposure levels in rooms and to sound an alarm when specified levels are exceeded. Also, some air exhaust monitors and liquid effluent

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monitors use GM tubes for monitoring low level contamination.

5.4 SUMMARY

Geiger-Mueller counters are used primarily for measurement of alpha and beta particles and x- and gamma-ray radiations. Their principal uses are for contamination and radiation field surveys and for general radioactivity counting. They can be fabricated in a wide variety of shapes but the standard geometry is cylindrical with coaxial electrodes, where the central anode wire is axially positioned and an end or side window is provided to admit ionizing radiation. The most common fill gas for the GM counter is one of the noble gases: helium, argon, or neon. Their use is to assure minimum electron attachment. Small amounts of a molecular gas (organic or halogen) are added to the major rare gas component to stabilize the avalanche discharge and prevent after-pulsing. The organic quenched chambers exhibit flatter counting rate plateau performance curves but have shorter useful lifetimes than the halide quenched tubes.

Their use of higher voltages and thereby higher levels of gas multiplication, provides them with a large output signal (approximately 1 to 10 volts) that can be easily processed without further amplification. Their output pulse height is constant and entirely independent of the energy deposited by the ionizing radiation. The release of a single electron-ion pair within the detector will result in a full amplitude output signal pulse. Thus, it is not possible to use differences in pulse height to discriminate among different particle types or energy distributions. However, this high degree of sensitivity makes the GM counter ideally suited for detecting fast electrons and photons because the small ionization of these particles would be hard to detect otherwise.

GM counters exhibit relatively large dead-time counting losses due to space charge deactivation of the gas multiplication process for a period following an avalanche discharge. Characteristic dead-times are typically 100 to 200 microsec. Dead-time losses limit the usefulness of GM counters, causing deviations from counting rate linearity with increasing source intensity and becoming appreciable for counting rates greater than a few thousand cpm, typical of fields of about 50 to 100 mR/hr. For higher exposure rates the GM counter saturates and eventually becomes blocked. Care must be exercised in their high-field use (i.e., with backup ion chambers) in order not to misinterpret saturation losses and/or blockage with that of a decreasing field intensity and/or zero field, respectively.

Their great utility is the results of several of the factors mentioned above, including high sensitivity, stability of operation, reliability of performance, simplicity of construction, ruggedness and relatively low cost for the GM tube and its electronics. Their principle disadvantages are associated with their dead-time loss characteristics and their lack of absorbed dose and energy dependent measurement capabilities.

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CHAPTER 6

SCINTILLATION DETECTORS

G. S. ROESSLER

6.1 PRINCIPLES OF OPERATION

Scintillation counting, one of the earliest methods known for measuring radiation, dates back to W. Crookes in 1902 (Crookes, 1903). He observed flashes of light caused by radiation interacting with a scintillating material, by using a "spintariscopes," which consisted of a small tube with a lens at one end and a zinc sulfide screen at the other. An alpha source was placed between screen and lens. Observation of the screen through the lens in a darkened room revealed flashes of light on various parts of the screen.

This detector did not gain widespread use outside the laboratory since it required a human "counter" to record the flashes. It wasn't until photomultiplier tubes were developed that scintillation detectors became practical for routine radiation measurements. Also, the discovery of new scintillating materials such as anthracene and the alkali halides made better instruments possible, because these materials are transparent to the scintillation light they produce and therefore can be used in bulk rather than just as surfaces.

6.1.1 Theory of Operation

When ionizing radiation interacts with a material capable of scintillating the energy of the incoming radiation particle is expended to produce ionization and excitation. Eventually the ion formed recombines or the excitation decays resulting in light emission. Consequently a portion of the original radiation energy is re-emitted in the form of visible light. For a scintillator, the number of light photons emitted is proportional, on the average, to the amount of energy originally deposited. A photomultiplier tube is used to convert the scintillation photons to an electrical pulse which is counted or otherwise analyzed electronically. Commonly the scintillation material is a solid or liquid. However, there are also gaseous scintillators.

SCINTILLATION DETECTORS

6.1.2 Types or Classes of Scintillation Detectors

Scintillation materials are obtainable in a variety of forms such as inorganic crystals, organic crystals, liquid organic solutions, liquid inorganic solutions, solid organic compounds (plastics), glasses, and noble gases. However, the most commonly used materials are inorganic crystals, primarily alkali halides containing small amounts of impurities as activators for the scintillation process, and liquid organic solutions. The inorganic crystals are utilized primarily for the external sample procedure, while liquid organic scintillators are usually used for the internal sample method (see 6.2.1).

Substances are useful as scintillators if the energy absorbed is re-emitted as visible or near ultra-violet radiation. If there are allowed transitions between the excited state populated by the incoming radiation and the ground state of the scintillation material, the resulting light emission occurs within the lifetime of the excited state (typically between 10^{-8} sec and 10^{-10} sec).

There are preferred scintillators for various applications:

1. Alpha particle counting: scintillators used for the detection of alpha particles usually consist of a thin layer of finely powdered ZnS (silver activated). The layer need not be thicker than the range of the alpha particle in the material and should be as thin as possible to minimize absorption of the light produced. Frequently, the ZnS is coated on a plastic film. The coating is sealed and placed inside a light tight covering which is put in contact with the sample. The scintillator may be discarded after use. The light-tight covering must be kept as thin as possible to permit penetration of the alpha particle to the scintillator.

2. Beta particle counting: commonly used scintillators for beta-particle detectors are the organic crystals anthracene and stilbene, organic materials in liquid form, and also plastics incorporating organic scintillators.

In some instances, inorganic crystals are used. In particular calcium fluoride doped with europium; $\text{CaF}_2(\text{Eu})$, is an inorganic crystal that has replaced organic crystals (notably anthracene) in many beta counting applications. High Z materials are generally unsuitable for beta counting since their counting efficiency is significantly reduced by excessive scattering of incident beta particles away from the surface of the crystal.

Scintillators used in beta counting must be thicker than scintillators used for alpha counting, yet they still are usually only a fraction of an inch thick. They must be sufficiently thick for obtaining a reasonable pulse height from the beta particle being detected, but should be as thin as possible in order to minimize unwanted contributions from gamma-radiation and other highly penetrating ionizing radiation.

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Liquid scintillation counters, employing coincidence counting have gained wide acceptance for counting low energy beta particle emitters, such as ^{14}C , ^{35}S , and ^{45}Ca . The liquid scintillator method allows for samples to be placed directly in the medium. This is important for low energy radiation which might otherwise be absorbed in the light-tight cover of a solid scintillator. This method is also useful for air monitor filters, and for wiping materials.

3. Gamma counting: scintillators, for the detection of high energy photons include inorganic crystals, usually NaI(Tl) and occasionally CsI(Tl) , blocks of plastic scintillator, and containers of liquid solutions of organic scintillator.

Generally speaking, inorganic crystals are more expensive than organic scintillators, but are used where good energy resolution is required. Organic scintillators are used where energy resolution is not the prime concern and where it is desired to have large detector geometry and thus high geometrical efficiency at lowest cost.

4. Neutron counting: neutrons do not produce ionization in scintillation crystals, but are detected through their interaction with the nuclei of the scintillator. In plastic and organic liquid scintillators this interaction consists of collisions with hydrogen giving rise to recoil protons. Neutrons also interact with Li atoms in LiI(Eu) and other scintillators containing Li to produce an alpha particle and a triton. These energetic charged particles produce scintillations in the detector material.

Inorganic Materials

Inorganic scintillators are crystalline materials classified as insulators or semiconductors (Knoll, 1979). They are usually doped with impurities called activators to enhance their efficiency.

One sequence of events required for scintillation is summarized as follows:

1. A charged particle (or a gamma photon) passes through the detection medium forming a large number of electron-hole pairs created by the elevation of electrons from the valence to the conduction band.

2. The positive hole will quickly drift to the activator site and ionize it (the ionization energy of the activator will be less than that of the lattice site).

3. The free electron in the conduction band will quickly migrate through the crystal until it finds an ionized activator and then recombines with it forming an excited neutral atom.

4. The excited activator then relaxes to the ground state

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emitting a photon (visible) in the process. (See Figure 6.1.)

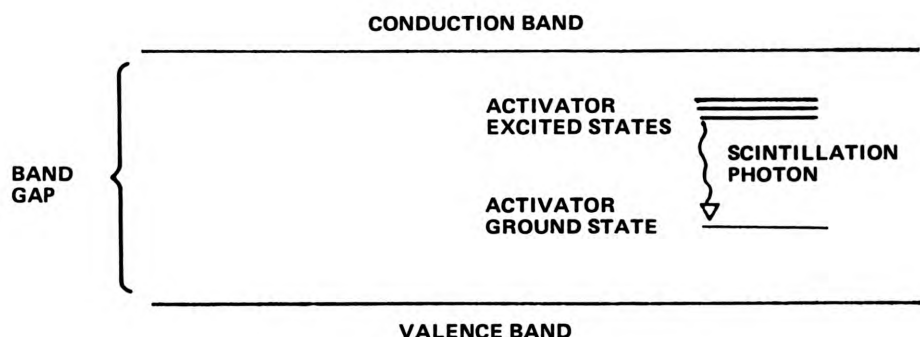


FIGURE 6.1 ENERGY STRUCTURE OF AN INORGANIC SCINTILLATOR.

In addition to this sequence there are other mechanisms that can lead to visible photon emission produced by the interaction of radiation with the scintillator (See Knoll, 1979). However, there are competing processes which do not lead to radiative transitions by the activators, these include:

1. Trapping of the electrons or holes in the lattice.
2. Quenching of the excited activator or its precursor (quenching here refers to all radiationless de-excitation processes such as the degradation of the "excitation energy" to heat.)

TABLE 6.1 CHARACTERISTICS OF COMMON INORGANIC SCINTILLATORS.

MATERIAL	WAVELENGTH OF MAXIMUM EMISSION (nm)	DECAY CONSTANT	DENSITY g-cm ⁻³	SCINTILLATION EFFICIENCY RELATIVE TO NaI(Tl)	REF.
NaI(Tl)	410	0.23	3.67	100%	d
CsI(Na)	420	0.63	4.51	85	d
CsI(Tl)	565	1.0	4.51	45	d
⁶ LiI(Eu)	470-485	1.4	4.08	35	d
ZnS(Ag) ^a	450	0.20	4.09	130 ^b	e
CaF ₂ (Eu)	435	0.9	3.19	50	d
Bi ₄ Ge ₃ O ₁₂	480	0.30	7.13	8	d
CsF	390	0.005	4.11	5	d
Li glass ^c	395	0.075	2.5	10	e

^a Polycrystalline.

^b For Alpha Particles.

^c Typical properties vary with exact formulation.

^d Scintillation phosphor catalog. The Harshaw Chemical Company.

^e Table of physical constants of scintillators, Nuclear Enterprises, Inc.

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Typical characteristics of inorganic crystal scintillators are: high density for good gamma ray stopping power and mechanical durability for fabrication purposes. The scintillation produced has a large pulse height good for detection of low energy interactions, and a short decay time for fast counting applications. Of these materials, thallium-activated sodium iodide crystals (NaI(Tl)) are especially good and are widely used for a variety of gamma-ray counting applications.

Sodium iodide has a density of 3.67 g/cm^3 , and the high atomic number of iodine ($Z=53$) is the reason why it is a very effective absorber for low and medium energy gamma rays. Also its response to energetic electrons is close to linear.

Table 6.1 summarizes some important characteristics of common inorganic scintillators.

Organic Materials

Scintillation of organic materials results from electronic transitions of the molecules. Organic compounds will fluoresce in any physical form; crystalline, liquid, or even vapor phase. Most organic scintillators are transparent to their own fluorescent emission.

Only two materials have gained widespread use as pure organic crystalline scintillators. One is anthracene which has a high scintillation efficiency, the other one is stilbene which has a somewhat lower scintillation efficiency. The latter one is preferred in situations where it is necessary to distinguish among scintillations induced by various particles and radiation. The disadvantage of both materials is that they are relatively fragile and difficult to prepare in large sizes (Knoll, 1979).

A large number of liquid organic solutions are available. These are produced by dissolving an organic scintillator in an appropriate solvent. Sometimes a wavelength shifter is added to transform the wavelength of the light into one which can be easily detected by the photomultiplier tube. Alkylbenzenes such as toluene and xylene are most often used as the solvent. The primary fluor used most often is 2,5 - diphenyl oxazole (PPO). A slightly more efficient scintillator is 2-phenyl-5-(4 - biphenyl) -1,3,4-oxadiazole (PBD). However its higher cost has prevented its widespread use. The wavelength shifter most often used is 1,4-bis-(2 - (5-phenyloxazolyl))-benzene (POPOP).

Polymerization of an organic scintillator solution produces a plastic scintillator (Knoll, 1979). An example is a styrene monomer in which an appropriate organic scintillator is dissolved. The solid plastic is then formed by a polymerization process. The main advantages of plastic scintillators are that they are inexpensive and can be fabricated into a variety of shapes.

Thin films of a plastic scintillator provide a detector which is thin compared to the range of even weakly penetrating particles such as heavy ions. These films act as transmission detectors which

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respond to only the fraction of energy lost by the particle as it passes through the detector. However this technology is still in the research stage. Some researchers have been able to make practical thin detectors of this type (Muga et al., 1972). Others report that the light yield is not linear nor is it related to the deposited radiation energy. Consequently this type of detector may or may not lend itself to practical usage (Braun-Munzinger and Gelbke, 1974).

Loaded organic scintillators are materials to which high Z elements have been added. The result is an increase in the photoelectric cross section of these materials. This increases the efficiency of the detector (i.e., its sensitivity) to higher energy photon emissions. Unfortunately this also leads to a decreased light output. An example of such a scintillator is the addition of gadolinium to a liquid organic scintillator. The high capture cross section for neutrons produces gamma rays which are detected directly in the scintillator.

Gas Scintillators

The first gas scintillation detector was constructed several decades ago (Grun and Schopper, 1951). It consisted of a chamber containing argon gas at atmospheric pressure with a small nitrogen impurity. Later investigations showed that xenon produced the highest light yield for this type of detector (Nobles, 1956). The light output can be increased by application of an electric field. More recently, a transparent parallel plate gas scintillation proportional counter has been designed which provides good resolution and efficient use of counter volume (Palmer and Braby, 1974).

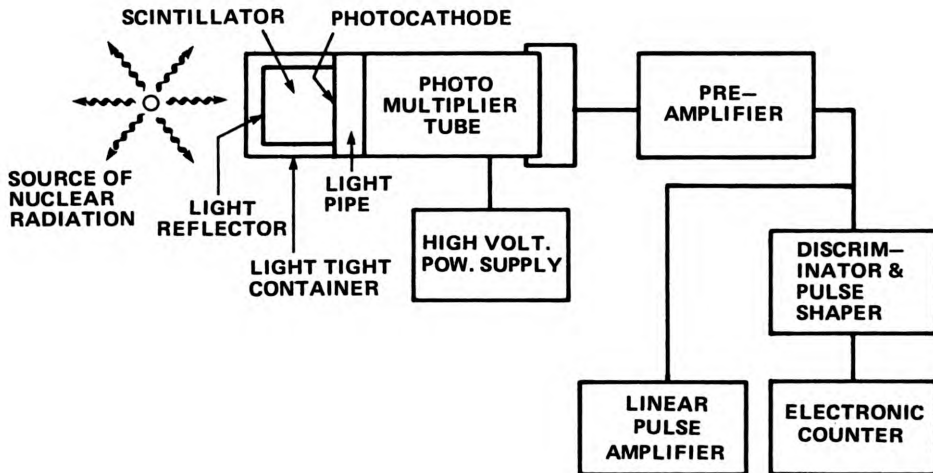


FIGURE 6.2 SCHEMATIC DIAGRAM OF AN INORGANIC CRYSTAL-TYPE DETECTOR.

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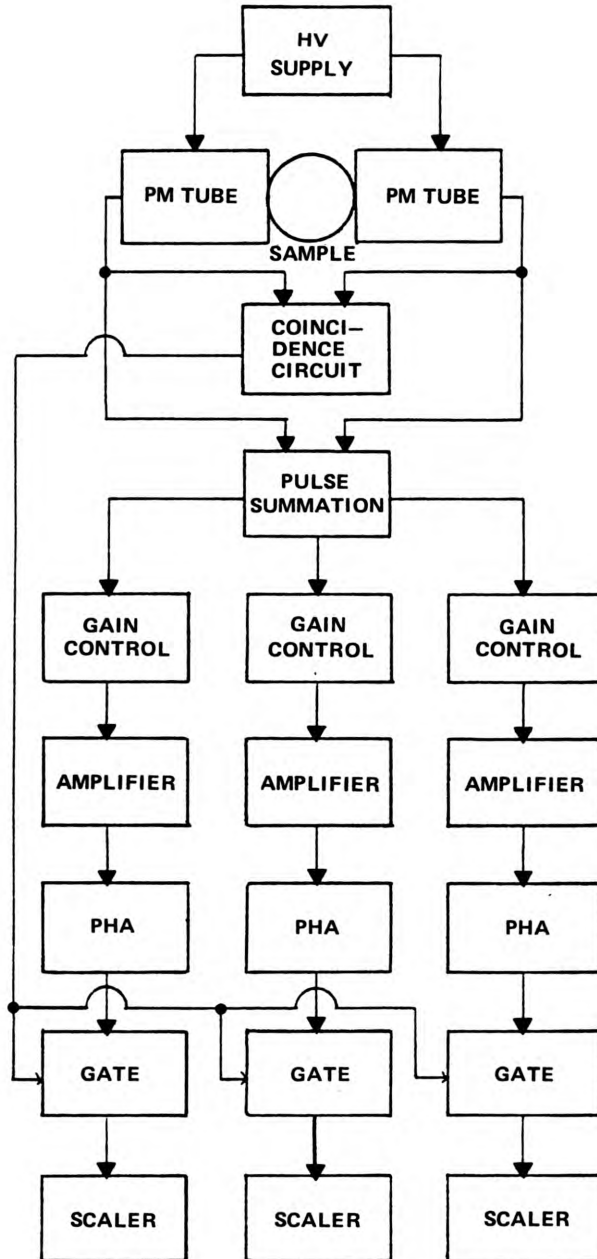


FIGURE 6.3 LIQUID SCINTILLATION COUNTING SYSTEM.

6.1.3 Instrumentation

Detector systems used for both external and internal sample measurements utilize photomultiplier tubes to convert scintillations produced by ionizing radiation in the detector

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material into a voltage output pulse. The schematic diagram for an inorganic crystal type detector system is outlined in Figure 6.2, while Figure 6.3 is a typical diagram for a liquid scintillation counting system.

A typical photomultiplier (PM) is a vacuum tube that contains a "photocathode," a series of electrodes (often 10 to 12) called "dynodes," and a final electrode, the anode (see Figure 6.4). When high voltage (800 to 1500 volts) is properly applied to the PM tube the "cathode" is negative and each successive dynode down to the anode is at a more positive potential, the anode being the most positive.

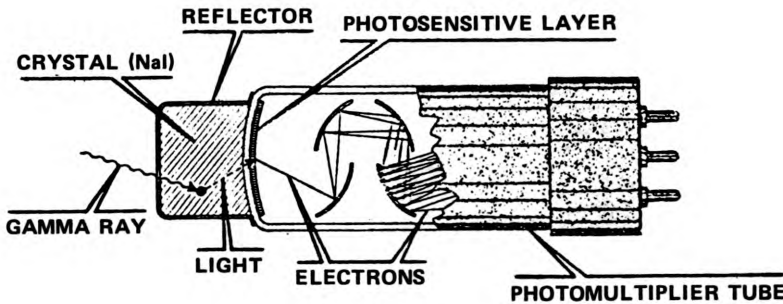


FIGURE 6.4 CRYSTAL PHOTOMULTIPLIER ASSEMBLY ILLUSTRATING THE DETECTION OF A GAMMA RAY BY THE CRYSTAL AND ITS SUBSEQUENT AMPLIFICATION THROUGH THE PHOTOMULTIPLIER TUBE.

When an incident photon from the scintillation strikes the photocathode, an electron is ejected and accelerated by the positive potential toward the first dynode. If it strikes the dynode with sufficient energy, a number of electrons will be emitted and they will tend to be accelerated toward the next dynode where more electrons are emitted. The process of multiplication continues down the dynode chain and the electrons are finally collected at the anode. If a tube has ten stages and an average multiplication of six per stage, an initial single incident light photon will result in 6^{10} or 6×10^7 electrons at the anode. The actual multiplication or "gain" of the PM tube is a function of the materials and construction of the tube itself and is highly dependent on the high voltage applied to the tube. Ambient light striking the photocathode of the PM tube would cause background counts unrelated to the radiation being measured and excessive light reaching the photocathode when the high voltage is applied to the PM tube may permanently damage the tube. The scintillator and the PM tube must be in optical communication with each other, and must be contained in a light-tight enclosure to exclude ambient light.

With the proper circuitry, the "burst" of electrons at the anode appears as an electrical impulse that can be amplified, transmitted, and further processed electronically or counted. With

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proper collection of light the magnitude of this pulse is proportional to the amount of energy deposited in the scintillator by the initial radiation.

Usually a preamplifier is located near the PM tube, often in the same housing. The preamplifier serves to match the anode output impedance to that of the cable so that the pulse is not seriously degraded in transmission to the analyzing instrumentation and it may also amplify (increase the magnitude of) the pulse. Scintillation counting instrumentation usually consists of a main amplifier of variable gain and either 1) a discriminator and counter (to record the number of pulses whose magnitude exceeds the level of the discriminator) or 2) some form of pulse height analyzer to record pulses or counts as a function of pulse height (i.e., as a function of energy deposited in the detector).

6.2 NORMAL OPERATION USE

6.2.1 Typical Configurations

Scintillation detectors are used in two general configurations:

1. External sample: the sample is located separately from the scintillator. The radiation to be detected must be able to pass through the scintillator's light-tight covering.

2. Internal sample: the sample and the scintillator are mixed or dispersed in the same container inside a light-tight covering

External Sample Scintillation Detector

Scintillation detectors using solid scintillation material may be used either as simple counters of, or as energy spectrometers of the ionizing radiation incident upon the detector. When used as counters their application is much the same as Geiger-Muller or proportional detectors, namely, to establish the occurrence of a radiation event. A detector-spectrometer system records not only the number of occurrences of an ionizing event, but also the amount of energy deposited by each radiation event, which is hopefully proportional to the energy of the radiation event itself.

In the external sample scintillation detector, the scintillator is optically coupled directly to the PM tube. The scintillator material is contained in an assembly which has a window facing the PM tube; or the scintillator material (or the assembly window) is coupled with fiber optics to the PM tube. Hygroscopic solid scintillators must be hermetically sealed in a container which then necessitates a window, while liquid scintillators can be coupled

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either directly or through a window to the PM tube.

Typically scintillation crystals are usually in the form of cylinders. Laboratory scintillators range in size from 1 x 1 inch to 4 x 4 inches; 3 x 3 inches being a common size. Special applications may employ other sizes and configurations. At times crystals 4 feet thick x 9 feet in diameter or larger have been used. Counting efficiency may be improved by a design which has a well in the center of the crystal. A test tube containing the radioactive solution is inserted into this well. In this way the sample is nearly surrounded by the scintillator.

The collection efficiency of the light output depends on geometry as well as on the crystal preparation. Sodium iodide is highly hygroscopic and humidity causes a yellow surface discoloration, which is probably due to free iodine. Crystals used as scintillators have to be transparent and free from discoloration. Therefore it is important to prepare the crystal in a dry atmosphere and to keep it in such an environment. For this reason the crystal is encased in a metal container which is hermetically sealed and frequently mounted directly on the phototube. In some cases the crystal is coupled through an intermediate window to the phototube face plate. The intermediate window material is typically quartz or glass. Such intermediate optical windows are used either to enhance uniform light collection or to provide for the removal of the phototube from the scintillator without exposing the crystal to moisture.

The phoswich (phosphor sandwich) detector is used to count alpha-beta radiations or low energy x- and gamma radiations. In the first case a thin alpha-beta scintillation crystal, such as CaF_2 , is coupled to a thicker, high density crystal such as NaI(Tl) . The thin crystal preferentially absorbs the low energy radiations. In the case of the low energy photons, CsI(Na) is used as the primary detector and the NaI(Tl) is used to detect the high energy interactions.

Such a detector is designed to discriminate against high energy interactions which would be due to sources other than the radiations to be measured. Compton scattering of high energy photons in the thin crystal will appear as background. Therefore, when a high energy photoelectric event is observed in the thick detector, the corresponding Compton interactions can be subtracted from the pulses recorded in the thin detector. The difference in the decay times of the two scintillators provides the mechanism for discriminating between the interactions. Anticoincidence circuitry provides the means for separation of the pulses.

Plastic scintillators may be machined in the same sizes and shapes just described for crystals. In addition, large blocks of plastic or tanks of liquid equipped with multiple PM tubes may be used to achieve a good geometry by partially or totally surrounding large samples, live animals, or even human beings (total body counters).

The use of a solid scintillation detector as a counter or a survey meter, although not providing any information about the energy

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of the radiation, has certain advantages (Wang et al., 1975). The main advantage is increased efficiency. Other advantages are ruggedness and minimal signal processing requirements.

Internal Sample Scintillation Detector

Liquid scintillators are useful for measurement of radioactivity in samples which can be put into solution with a scintillator. Pure beta emitters such as ^{14}C , ^3H , and ^{35}S , can be best analyzed in this way. Liquid scintillation counting is based on the fact that the amount of light produced is proportional to the energy of the beta particle. The proportionality of response makes energy discrimination and pulse height analysis possible. However, the continuous nature of the beta spectrum (e.g., Figure 6.5) or the Compton spectrum limits this method in regard to energy resolution. However, in liquid scintillation counting energy resolution is secondary. The main advantage is the great sensitivity of this method in low-energy beta detection and certain applications for the detection of gamma rays.

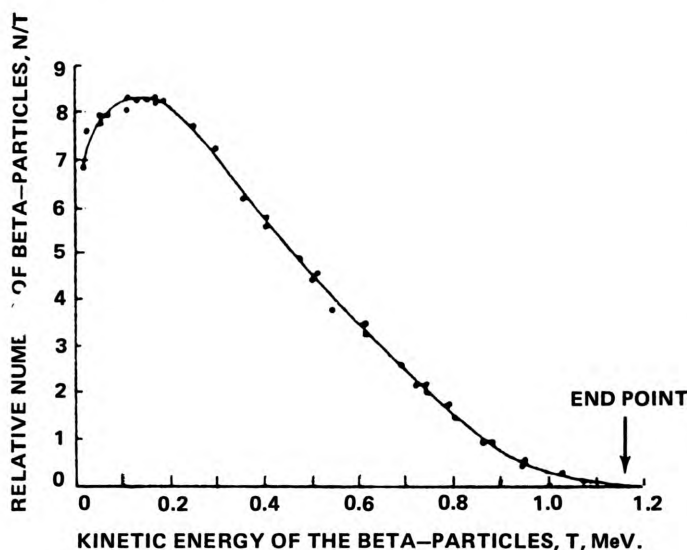


FIGURE 6.5 BETA-PARTICLE SPECTRUM OF RaE.

A typical liquid scintillation counting system consists of a scintillation "cocktail" (liquid scintillator plus internal source) viewed by two opposed photomultipliers operated in coincidence (see Figure 6.3), a temperature-control system, electronic circuitry to process the pulses, and a heavy shield against gammas.

Since liquid scintillators emit blue light, the photomultiplier tubes used for liquid scintillation counting must have a high sensitivity in the blue region of the spectrum. Photomultiplier tubes using bialkali photocathodes (K_2CsSb) have excellent sensitivity in this region.

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Commercial systems incorporating all of the above components are available. They handle up to 400 samples automatically and provide output in two to five energy channels (windows).

The photomultiplier tubes are usually cooled using commercial (freezer type) refrigeration units or thermoelectric cooling in order to reduce the thermal noise coming from the photomultiplier tube. Operation of the system with a coincidence circuit is also used for reducing noise.

The simplest mode of operation is to feed the output pulse from the coincidence circuit directly into a scaler. A more advanced approach is to use the pulse height analysis mode instead of the single scaler mode. However most liquid scintillation systems are used as spectrometers with three to six discriminators. The low energy discriminator is set to reject pulses below a certain amplitude, a measure intended for noise reduction. The other discriminators are used to measure (in conjunction with the lower discriminator) two or more energy regions.

The vial used for the liquid scintillator and sample is made of a special glass having a low potassium content, since natural potassium contains ^{40}K , which is radioactive.

Quenching is a process which reduces the intensity of the flashes of light produced by the beta particles in the scintillator. This must be considered in the calibration. Quenching can be thermal, which comes about if the solvent converts the beta energy into heat (e.g., radiative transitions in the material in favor of non-radiative (collisional) transitions). Color quenching occurs if the solvent absorbs the light emitted by the scintillator.

The optimum combination of solvent, scintillator, counting vials, and counting conditions for a particular sample can best be determined by following recommendations of the manufacturers of the particular instrument used. However, operational procedures such as calibration and determination of counting efficiencies are fairly common to all systems.

The instrument is either automatically or manually operated. Generally, liquid scintillation samples fall into only two categories: 1) samples with unknown or varying quenching, and 2) samples of known or constant quenching. An automatic system has to be programmed to take care of the various measurement parameters. Under automatic operation, all samples are loaded into the system sample holders, the proper parameters for counting are selected, and the sequence started. The user may return later or the next day to pick up the printout. In the manual mode, individual samples may be run by the operator with different instructions for each.

6.2.2 Spectrometry

A scintillation spectrometer, like any other spectrometer, is an instrument for measurement of the energy distribution of radiation. It has widespread application in both identification and

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quantification of radionuclides in samples.

A complete discussion of the interactions of radiation with matter, photoelectric absorption, Compton scattering, and the relation of pair production to photon spectroscopy, along with the production of spectra, is beyond the scope of this text. The reader is referred to texts devoted to this field (Birks, 1953; Wang et al., 1975).

Here it must suffice to state that the photopeak is used as the means of determining photon energy with inorganic detectors. The Compton continuum or the beta spectrum is used, although yielding a less accurate result, to determine photon or beta energy with organic detectors.

6.3 CALIBRATION

The calibration of any radiation measuring device simply means that the instrument is capable of reproducing a standardized response indicative of the amount of energy or number of energy releasing events involving radiation incident on its sensitive region. With scintillation detectors, as with any other radiation instruments, one is interested in determining accuracy and precision of the instrument. Accuracy refers to the degree of agreement between the observed value and the correct value. It answers the question, "How close are you." Precision or reproduceability refers to the degree of repeatability of the measurements made by the instrument.

This chapter has discussed all types of scintillation instrumentation together because the basic detection principles are the same. Calibration techniques, however, fall into a number of distinct categories and these will be discussed separately.

6.3.1 Survey-type Scintillation Detectors

The scintillation type survey meter is calibrated in much the same manner as other types of survey meters already discussed. The calibration must be carried out for various radiation types and energies (most instruments are energy dependent). The calibration is performed by placing the instrument in a position in a radiation field where the dose rate is known. The dose rate at the point of measurement can be calculated from the known activity of a standard source (usually a secondary or a substitution standard) by using the dose constants listed in Table 6.2 (Kiefer and Maushart, 1972) and the inverse square law of distance for large source to detector distances. Dose rate can also be determined by the use of an instrument which has been calibrated by comparison with a standard chamber that can perform an absolute measurement of the dose rate.

Prerequisites include a calibrated ^{137}Cs or ^{60}Co gamma source, and a calibrated ^{239}Pu alpha source. The instrument should

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be calibrated at the 2/3 full scale meter position on each scale and linearity should be checked at the 1/3 full scale position on each scale, following the principles outlined in Chapter 2.

TABLE 6.2 SPECIFIC GAMMA-RAY EMISSION, I_γ OF VARIOUS RADIONUCLIDES (KIEFFER AND MAUSHART, 1972).

Nuclide	Approximate quantum energy of γ -radiation (MeV)	Dose contribution in %	I_γ , in $\frac{R}{h} \times \frac{m^2}{c}$
^{22}Na	1.28	100	1.26
^{24}Na	1.380	38.6	1.89
	2.760	61.4	
	(eff. 2.07)		
^{52}Mn	0.73	100	1.92
	0.94	100	
	1.46	100	
^{54}Mn	0.81	100	0.45
^{59}Fe	0.191	4.5	0.65
	1.098	53.5	
	1.289	46.0	
	(eff. 1.20)		
^{58}Co	0.81	100	0.56
^{60}Co	1.172	47.5	1.35
	1.332	52.5	
	(eff. 1.25)		
^{64}Cu			0.114
^{65}Zn	1.12	47	0.30
^{72}Ga	(eff. 1.38)		1.6
^{82}Br	0.547	21.8	1.5
	0.787	30.6	
	1.35	47.5	
	1.7-2.0	0.1	
	(eff. 0.893)		
^{130}I	0.417	40	1.23
	0.537	100	
	0.667	100	
	0.744	100	
^{131}I	0.080	6.3	0.231
	0.284	6.3	
	0.364	80.9	
	0.637	9.3	
	0.722	2.8	
^{137}Cs	0.661	100	0.34
^{152}Eu	83% on the average 1 MeV 17% between 200 and 400 keV		~0.7
^{182}Ta	(eff. 1.13)		0.61
^{182}Ir	(eff. 0.60)		0.61
^{198}Au	0.423	~100	0.24
^{203}Hg	0.280	~100	0.16
^{226}Ra (0.5 mm Pt filter)	(eff. 0.84)		0.825

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6.3.2 Single Channel Scintillation Spectrometers

The measurement of pulse amplitude is one of the most important aspects of counting and spectroscopy of radiation. The amplitude of the output pulse from a linear electronic amplification system will be proportional to the energy deposited in the detector. Pulse height analyzer (PHA) systems can be classified as either single or multichannel analyzers.

Figure 6.6 shows a complete single channel analyzer (SCA) system. Such a system is useful if one, or at most a few, gamma-ray spectral peaks are of interest. The SCA can be operated in the window mode so that upper-level and lower-level discriminator thresholds can be adjusted to encompass the expected pulse amplitudes corresponding to the energy of the photon of interest. Whenever a pulse from the pulse amplifier has an amplitude between the upper and lower-level threshold, a digital pulse is generated and sent to a recording instrument.

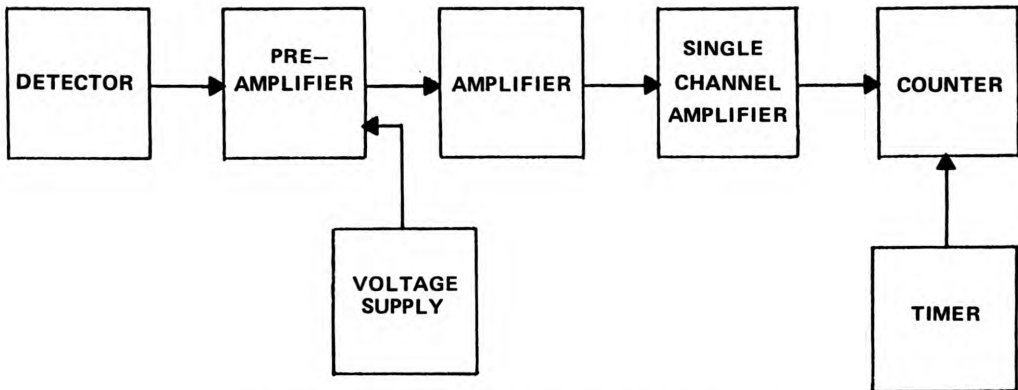


FIGURE 6.6 SINGLE-CHANNEL COUNTING SYSTEM.

A number of factors must be considered in the calibration of any detector system in which spectrometry plays the major role in the output of the radioactivity measurement. These factors, generally not considered in calibration of other types of instrumentation, include resolution, pulse height analysis, and data reduction. Methods for evaluation of these factors will be covered in this chapter, but also apply in Chapter 7, "Semiconductor Detectors." In liquid scintillation in particular, counting calibration involves consideration of geometry, energy and efficiency.

Geometry

The size and position of the radiation source with respect to the detector is an important criterion in determining the sensitivity and the energy of response of a particular system.

The solid angle subtended by the detector with respect to the source is a function of the distance of the source from the detector

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and the size of the detector's sensitive region. The size of this angle, and the position of the source determines the fraction of emitted photons or particles that will be incident on the detector (see Figure 6.7).

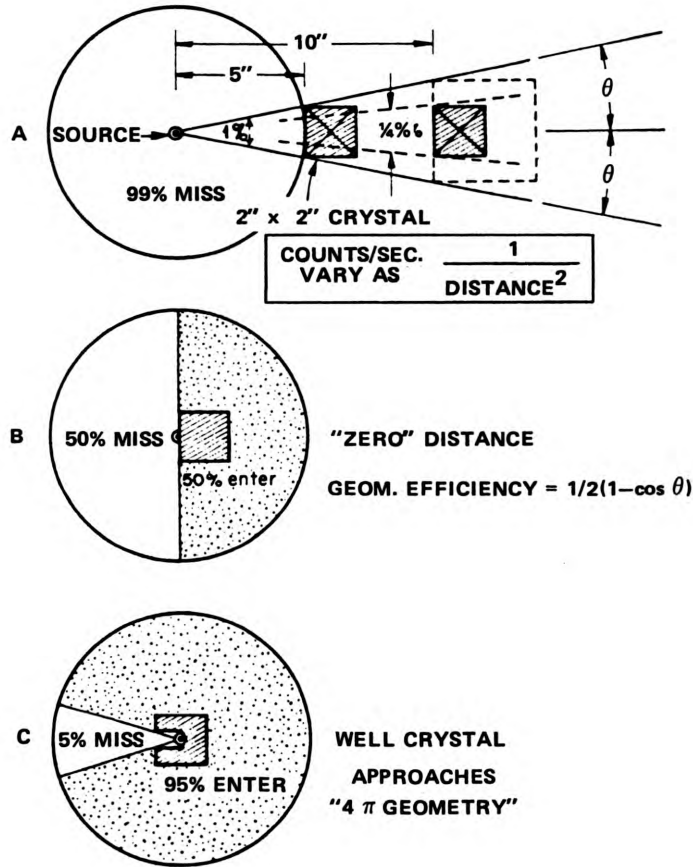


FIGURE 6.7 THREE STAGES OF DETECTOR GEOMETRY. A, WITH THE SOURCE 2½ CRYSTAL DIAMETERS AWAY, THE GEOMETRY IS DOWN TO 1 PER CENT. DOUBLE THE DISTANCE AND IT BECOMES 0.25 PER CENT—INVERSE-SQUARE LAW. B, WITH THE SOURCE AGAINST THE CRYSTAL THE GEOMETRY WILL BE 50 PER CENT—COSINE LAW. C, WITH A WELL CRYSTAL THE GEOMETRY APPROACHES 100 PER CENT (4 π), BUT THE SAMPLE MUST BE SMALL (AFTER WAGNER, 1968).

If the detector is placed in a homogeneous photon field, the directional dependence of the detector must be understood. Thus a knowledge of the shape of the detector, in addition to any previous calibration information showing geometries used, must be available. For discrete sources of radiation, such as calibration standards, the source should approach the point-source geometry to assure almost inverse square adherence in the intensity versus distance criteria at large source to detector distances.

The size and position of the sample or radiation source with

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respect to the detector is, as in most detector systems, of utmost importance in determining the efficiency and the energy response of the system. This means that the calibration source should be the same phase as the sample, either gaseous, liquid, or solid. For a given sample size (the sample should be as high in activity as practicable for maximum sensitivity), the sample should be distributed so as to minimize the distance between the sample volume and the sensitive region of the detector.

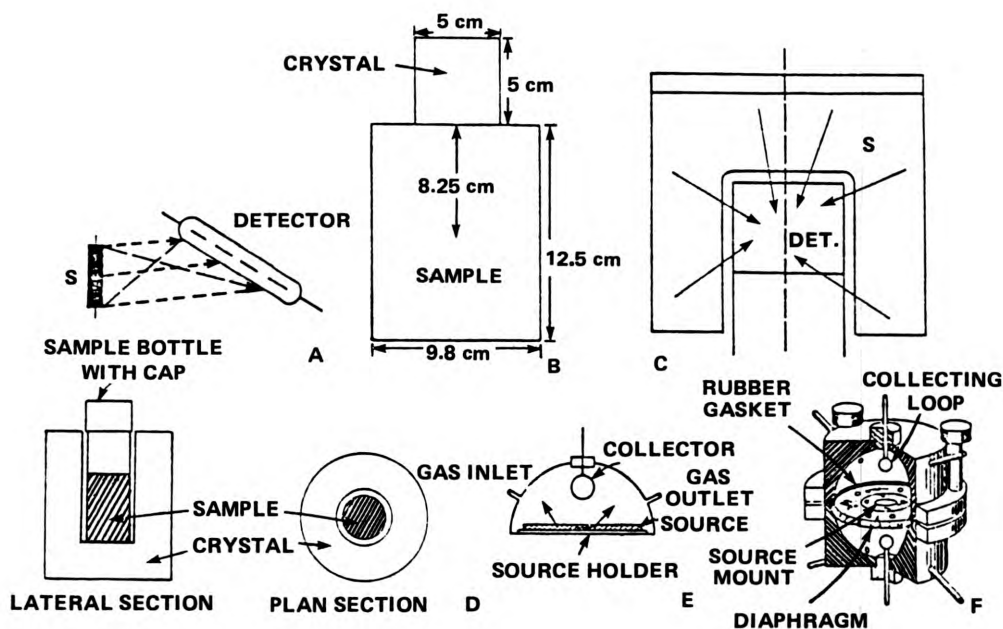


FIGURE 6.8 EXAMPLES OF VARIOUS SOURCE (S)—DETECTOR GEOMETRIES: (A) POOR GEOMETRY; (B) SMALL SOLID ANGLE; (C) MARINELLI BEAKER SURROUNDING DETECTOR; (D) WELL-COUNTER (NEARLY 4π); (E) WINDOWLESS FLOW COUNTER WITH 2π GEOMETRY; AND (F) 4π GEOMETRY COUNTER FOR ABSOLUTE BETA DETERMINATIONS.

The activity of a source will also determine its position with respect to the detector. Low activity sources should be placed close to the detector for maximum sensitivity. High activity sources will increase the dead-time in a system and may affect resolution, so placement at a distance may be necessary.

Geometry considerations in liquid scintillation counting of beta-emitting nuclides are unique since the sample is incorporated into the scintillation solution. The best geometry or contact between the sample and the fluor is achieved when both are dissolved in the same solvent mixture. A number of techniques for achieving this have been compiled (Wang et al., 1975). Radioactive samples that cannot be dissolved in some suitable solvent system may still be assayed by the liquid scintillation process in a heterogeneous system. The sample may be in the form of a gel or an emulsion, or they may be dried on filter paper and the paper strip placed in a fluor

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solution (Rapkin, 1961). Aqueous samples may be introduced into a counting vial containing a solid, insoluble fluor such as anthracene.

In any case, geometry plays a part in the accurate determination of other factors such as efficiency and energy calibration. Generally speaking, for adequate analysis of a given sample the detector should be calibrated with a source of the same physical form. Figure 6.8 shows examples of a number of source-detector geometries which are commonly used in scintillation counting.

Energy Calibration

Since scintillation counting is a proportional response system (i.e., the magnitude of the output signal from the detector is proportional to the energy dissipated by the radiation particle in the scintillator), it is necessary to optimize the instrument high voltage, amplifier gain, and window discriminator settings. For instruments with more than one window or channel, each window must be optimized for a particular counting situation.

Most commercially available units have three or four single channel analyzers so that the activities in a sample with up to four different energies can be measured simultaneously.

Energy calibration for gamma-emitters can be achieved by the following steps:

1. Select an energy range e.g., 0-500, 0-1000 keV.
2. Determine the first point of curve:
 - a) Select a secondary calibration source, such as ^{137}Cs or ^{60}Co .
 - b) Set HV to a point in the operating range of the photomultiplier tube (800-1000 volts for many tubes).
 - c) Select a window (ΔE) width (e.g., 10% of baseline full scale).
 - d) Set baseline (E) to center window around desired location for photopeak of the selected source in the proposed calibration range.
 - e) Adjust Coarse Amplifier (Gain or Attenuator) setting to put peak in the desired window (adjust to maximum count rate).
 - f) Reduce window width (ΔE) (such as to 1% or 5%).
 - g) Make fine gain adjustment (maximize the count in the narrow window) (i) with amplifier if it has a fine control, or (ii) with fine HV adjustment. Do not exceed HV limit of the tube.
3. Check linearity and intercept:
 - a) Use additional sources of other energies in range of interest and without changing systems gain (or attenuator) or window width (ΔE) to find locations of the peaks by varying baseline control (E). Peak value corresponds to baseline + $1/2$ the window width ($E = 1/2 \Delta E$).

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b) By inspection or by plotting on graph paper determine if energy vs. dial setting curve is linear with a zero intercept. If not, instrument must be adjusted or the calibration curve must be used in interpreting energy values of dial settings.

Calibration for beta-emitters follows much the same procedure except ^3H , ^{14}C , ^{35}S and ^{45}Ca sources would serve as secondary standards and window width would be larger due to the fact that the window must be centered on the maximum of the beta-energy spectrum rather than a well defined photo peak.

Efficiency Calibration

The efficiency of a liquid scintillation counter depends on several factors including temperature, background count rate, and quenching.

Temperature is controllable by most systems so that thermal noise is kept to a minimum. Background count rate also is controlled by a properly shielded detector and by an optimum setting of the windows for a maximum peak to background ratio for any radionuclide.

Quenching has by far the greatest effect on counting efficiency since it limits the amount of sample that can effectively be incorporated into the solution. The introduction of the sample almost always reduces the scintillation light output compared to the pure scintillator.

Due to the unpredictable nature of most samples the counting efficiency must be determined for essentially every sample because of the difference in composition and degree of quenching. Two methods may be used to determine efficiency. The older and more accurate method is the internal standard or spiking method. It compares the sample count rate, C , with that of the sample plus an added activity, S , of the same radionuclide.

Efficiency, ϵ , is calculated by:

$$\epsilon = \frac{(C + S) - C}{S} = \frac{S}{S} \quad (6.1)$$

where s is the observed increase in count rate due to the activity, S , of the added standard. For an accurate determination of C , s should be equal to or greater than C .

This method is very reliable and applicable to both chemically quenched or color quenched samples if applied correctly. However, this method is time-consuming in that two counting processes are required for each sample and the sample vial must be opened between the counts. The accuracy of the method may depend on accurate measurement of small volumes of the standard. Furthermore, the original sample is contaminated by the addition of the standard and cannot be counted again.

The second approach, the external standard method, is more convenient. In this situation, a sealed gamma source or combination

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of sources, such as ^{241}Am or ^{226}Ra , is incorporated into the system by the manufacturer and is moved from a shielded remote storage container into a position adjacent to the sample. The sample can be counted with and without the standard to measure quenching. This technique makes use of the Compton electrons derived from the interaction of the gamma rays from ^{241}Am or ^{226}Ra with the glass vial and the scintillation solution. The energy spectrum of the Compton electrons resembles the low energy beta-emitter spectrum and therefore provides a means of quantitating the amount of quenching in the sample. Correlation curves are run relating efficiency of any particular beta-emitting nuclide of interest to the count rate of the external standard.

Almost all liquid scintillation counters currently manufactured are equipped to perform counting efficiency determinations by this method.

The following step-by-step procedure can be used for calibrating most automated or manual liquid scintillation counting systems for two radionuclides of different beta particle energies. This procedure can be expanded to counting for three or more radionuclides by repeating steps 8-10 with other energy sources:

1. Set high voltage at minimum and turn instrument on.
2. Place a calibrated ^3H source in an appropriate scintillation-solvent solution into the sample holder and cycle the system until the sample is in the detector chamber.
3. (a) Set amplifier gains to maximum. (b) Set instrument to count in the coincidence mode. (c) Set window controls in wide mode.
4. Set lower discriminators to 100 units and upper discriminators to 1000 units (i.e., 1 volt - 10 volts).
5. Raise the high voltage in coarse steps (counting samples at each step) until count rate reaches a maximum and begins to decrease.
6. Set coarse high voltage at the position that gives maximum count rate and repeat step 5, using the fine high voltage control.
7. Calculate an ^3H efficiency in the following manner:
 - (a) Count the source for ten minutes, record.
 - (b) Count a background for ten minutes, record.
 - (c) Reduce both counts to counts per minute (cpm); subtract the background cpm from the source cpm to obtain a net count.
 - (d) Calculate the disintegrations per minute (dpm) by multiplying the source activity in μCi times 2.22×10^6 dpm/ μCi .
 - (e) Efficiency in percent is then:

$$\epsilon = \frac{\text{net cpm}}{\text{source dpm}} \times 100 \quad (6.2)$$

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8. Place a calibrated ^{14}C source in sample changer and cycle until the sample is in the detector chamber.

9. Count sample while lowering the amplifier gain on one of the channels until a maximum count rate is observed.

10. Count sample and background for ten minutes and calculate a ^{14}C efficiency as outlined in Step 7.

The Figure of Merit used in the quantitative evaluation of counting performance is ϵ^2/B (efficiency squared divided by background). The procedure outlined should achieve the best value. Note that it may be necessary to correct for deadtime losses and for source decay.

The instrument is now set up to give maximum counting efficiency for the particular sample-solvent scintillator used. Any changes, including different radionuclides, require re-calibration. On instruments with fixed high voltage it is necessary to use the amplifier gain to optimize count rate in each window setting.

Special Detectors

A special but less common type of radiation instrument is a whole or partial body counter which uses liquid scintillation as the detector. This detector has the same advantage of other liquid systems in that it has a high counting efficiency, but suffers in resolution of the gamma energies measured. The system relies on a good calibration of the Compton "peak" produced by interaction of the gammas with the low Z scintillation solution.

The sample is usually placed in the center of a right circular cylindrical tank containing the scintillator (see Figure 6.9). Therefore, calibration differs in several respects from that of the liquid systems already discussed. Much of the instrumentation, however, is very similar.

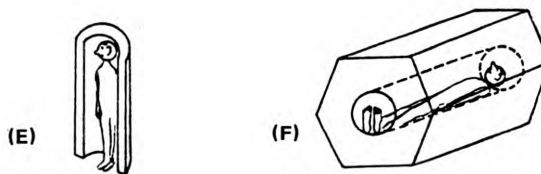


FIGURE 6.9 TWO CONFIGURATIONS OF WHOLE-BODY COUNTERS UTILIZING A LIQUID SCINTILLATOR AS THE DETECTOR.

With the exception of quenching considerations and the type of radiation emission involved, a whole or partial body counter is calibrated with consideration for the best geometry possible and following the same procedure as outlined for the internal standard method described earlier for energy and efficiency calibration.

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Another special detector which is more common is the partial or wholebody counter which uses one or more NaI(Tl) detectors. Several examples of these can be seen in Figure 6.10. The calibration of this type of detector can follow either the single channel spectrometer method already described or may often utilize the multichannel scintillation spectrometer (see the next section).

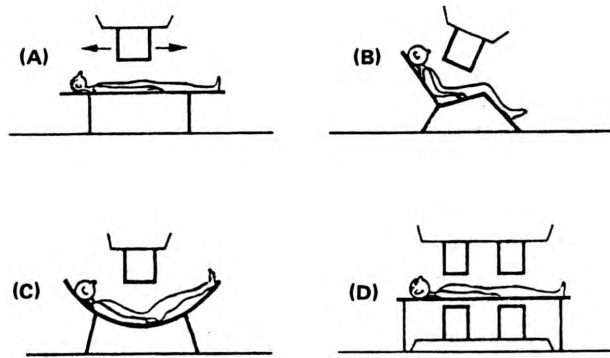


FIGURE 6.10 GENERAL CONFIGURATIONS OF PARTIAL OR WHOLE-BODY COUNTERS EMPLOYING NaI(Tl) DETECTORS.

6.3.3 Multichannel Scintillation Spectrometers

In applications where the radionuclide source emits photons of many different energies with different relative intensities, it is often necessary to use a multichannel PHA (MCA). Figure 6.11 shows a simplified block diagram of a MCA (in dotted lines) and some frequently used readout and storage systems.

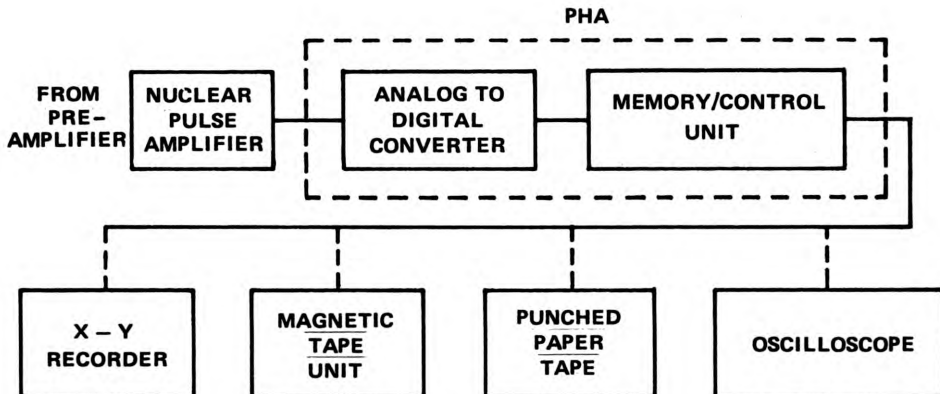


FIGURE 6.11 MULTICHANNEL PULSE-HEIGHT ANALYZER SYSTEM SHOWING VARIOUS READOUT AND PERMANENT DATA-STORAGE OPTION.

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The pulse-height analysis is performed by an analog-to-digital converter (ADC) and the result of the analysis of each pulse is stored in digital form in the memory.

In the calibration of detectors using a multichannel analyzer (MCA) system for the identification and quantification of radioactivity, one must consider all the factors (detector-source geometry, the system resolution, the system efficiency, and the data reduction-analysis), both independently and as a total system.

Geometry

Basic geometry considerations have been discussed in section 6.3.2. No special geometries apply specifically to systems using multichannel analysis approach.

System Resolution

One of the most important aspects of the scintillation detector system is the width of the energy peak resulting from the energy deposition in the scintillator. This is particularly important for the measurement of gamma-emitting nuclides. Since gamma rays are monoenergetic, theoretically a line spectrum would result. The actual spectrum, however, appears as shown in Figure 6.12.

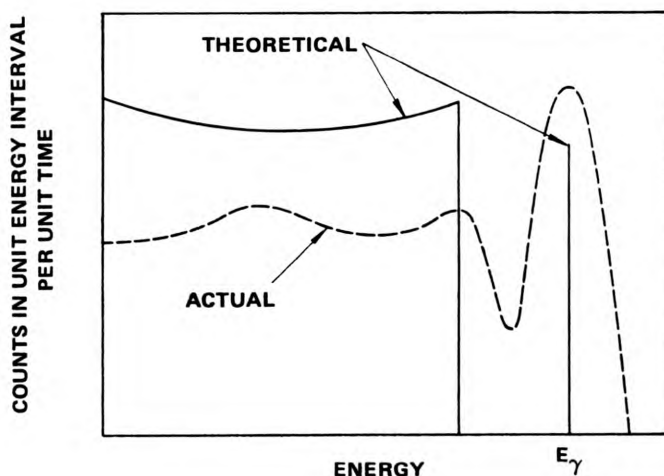


FIGURE 6.12 THEORETICAL AND ACTUAL ELECTRON-ENERGY DISTRIBUTIONS FOR SINGLE COMPTON AND PHOTOELECTRIC INTERACTIONS IN A 3" x 3" NaI(Tl) DETECTOR USING A 0.511-MeV PHOTON SOURCE. (FROM HEATH, 1964)

Compton interactions of the gamma radiation within the scintillator produce the lower energy portion of the spectrum and

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photoelectric interactions, multiple Compton and other interactions either singly or in combination, produce the high energy peak. The peak is "smeared" from a line to different widths depending on the particular detector's ability to resolve the gamma energy being counted.

The most common indicator of the resolution of a detector system is a measure of peak width - the "full width at half maximum" (FWHM), shown in Figure 6.13 where $\text{FWHM} = \Delta E_{1/2}$, as measured from the figure. One calculates the resolution by:

$$\text{Resolution} = \frac{\Delta E_{1/2}}{\Delta E} \times 100 \quad (6.3)$$

where $\Delta E_{1/2}$ is the number of energy units at the width of the peak at one-half the height of the peak, and E is the peak energy. The resolution can be calculated for each peak in an energy spectrum and will generally be different for each one.

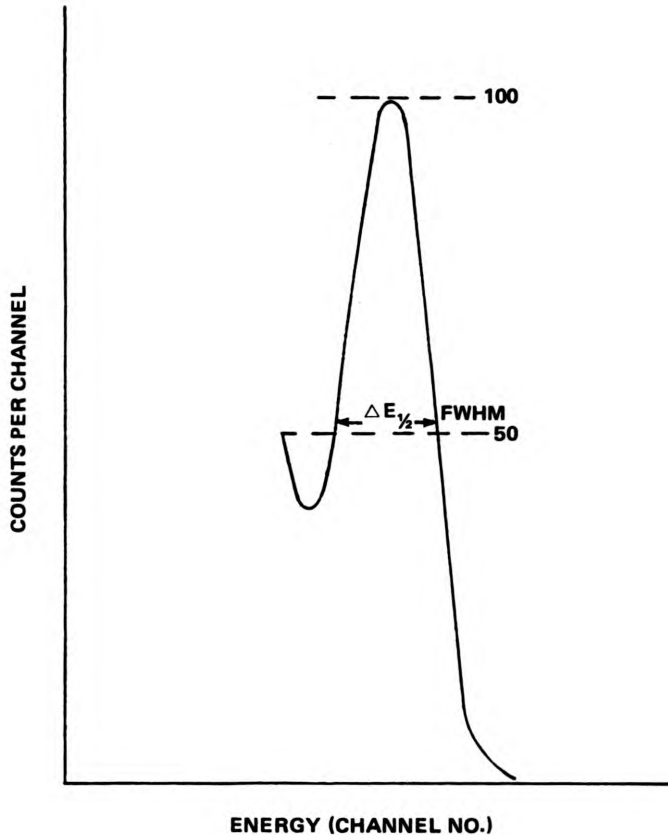


FIGURE 6.13 FULL WIDTH AT HALF MAXIMUM OF A PEAK.

The resolution depends on several factors including the number

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of optical photons that are released in the phosphor, the number that strike the photocathode, the number of photoelectrons released per photon hitting the photocathode, the number of photoelectrons that strike the first dynode, the multiplication factor per dynode stage and the number of dynodes characteristic of the photomultiplier tube.

In general, the greater the number of each of the above the better (i.e., the smaller) the resolution. The energy dependence of the resolution for scintillation detectors has been found to follow the empirical relationship:

$$R^2 = \alpha + \frac{\beta}{E} \quad (6.4)$$

where R is the resolution, and alpha and beta are empirical constants specific for the particular detector and photomultiplier tube, and E is the energy of the radiation being measured. R varies inversely as $E^{1/2}$. Therefore, the higher the E , the smaller (or better) the resolution. Resolution measurements on a system are generally made using radiation sources at specific known energies.

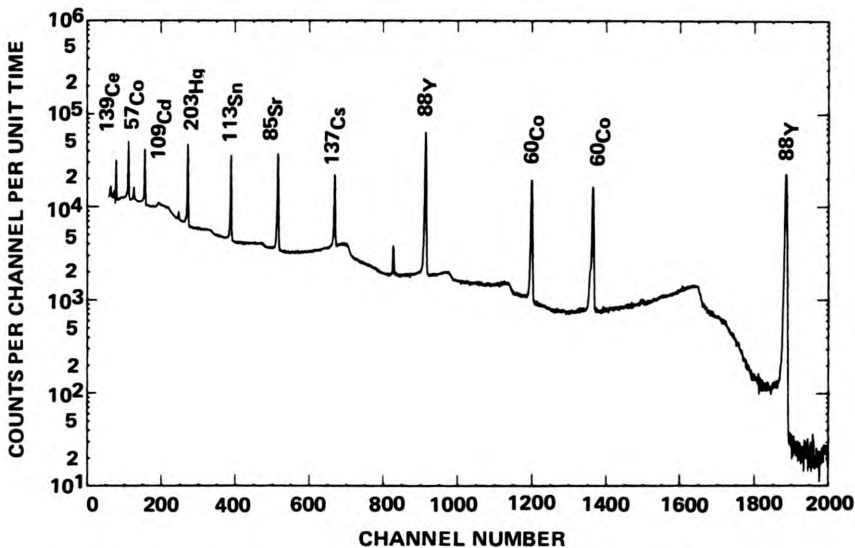


FIGURE 6.14 GAMMA-RAY SPECTRUM OF AN NBS MIXED-RADIONUCLIDE GAMMA-RAY EMISSION-RATE STANDARD (COURSEY, 1976).

The $\text{Ge}(\text{Li})$ gamma-spectra of a mixture of several radionuclides emitting numerous gamma rays of known energy (and spanning a wide range of energies) is shown in Figure 6.14. This example of a mixed radionuclide source is tabulated according to energy in Table 6.3. If this source is obtained from NBS, it can be considered a secondary standard and can be used to determine the system resolution as well as perform the energy and efficiency calibration of a system.

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TABLE 6.3 SECONDARY STANDARD FOR MULTICHANNEL SCINTILLATION SPECTROMETER
CALIBRATION — MIXED RADIONUCLIDE SOURCE.

PARENT RADIO- NUCLIDE	GAMMA-RAY ENERGY (MeV)	GAMMA RAYS PER DECAY USED	HALF LIFE	GAMMA-RAY EMISSION RATE ($\gamma \text{ s}^{-1}$) 1200 EST SEPT. 1, 1978
^{109}Cd	0.088	— — —	463.9 d	5.624×10^2
^{57}Co	0.122	0.8559 ± 0.0019	272.4 d	1.003×10^3
^{139}Ce	0.166	0.7994 ± 0.0013	137.7 d	6.029×10^2
^{203}Hg	0.279	0.815 ± 0.008	46.62 d	1.901×10^3
^{113}Sn	0.392	— — —	115.0 d	1.806×10^3
^{85}Sr	0.514	0.98 ± 0.01	64.85 d	3.009×10^3
^{137}Cs	0.662'	— — —	30.0 y	1.557×10^3
^{88}Y	0.898	0.950 ± 0.005	106.66 d	1.119×10^4
^{60}Co	1.173	0.9990 ± 0.0002	5.271 y	3.965×10^3
^{60}Co	1.332	0.99982 ± 0.00005	5.271 y	3.968×10^3
^{88}Y	1.836	0.9935 ± 0.0003	106.66 d	1.170×10^4

(Data taken from National Bureau of Standards certificate standard reference material 4215F.)

Energy Calibration

A secondary standard such as a mixed radionuclide source containing a number of radionuclides covering a wide range of energies (i.e. < 100 keV to 2 MeV) can be prepared in the various geometries which are used with a counting system. One can then associate with each photopeak in the pulse-height spectrum a channel number N (corresponding to the maximum point in the peak) and a gamma-energy, E , using the linear relationship where the constants a and b are determined graphically by measuring the spectra of gamma-emitters that have well known gamma peak energies: $E = aN + b$. This relationship can then be used to identify any peaks found in a sample spectrum by relating the number of the channel at which a peak is found to a particular energy. The calibration should be performed daily or more frequently, depending on the stability of the system.

Efficiency Calibration

The efficiency of a detection system is defined as the ratio of

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the number of particles or photons releasing energy to the detector and the total particle flux of the source. Factors influencing efficiency include the size and composition of the detectors and the source-detector geometric factor.

The efficiency in scintillation detectors is determined using sources of known energies and intensities. The type of source will depend on the radiation of interest: i.e., alphas, betas, photons, etc.

The efficiency of a detector for a given type of radiation and detector is often a function of the energy of the particle or photon.

The efficiency of a gamma scintillation spectrometer will also depend upon the particular portion of the characteristic gamma-ray spectrum that is used for the detection efficiency determination. It may be the full spectrum or only the peak.

Data Reduction and Analysis

The problems encountered in the analysis of pulse-height data from x-ray and gamma spectrometers require the application of sophisticated techniques to obtain the desired results (NCRP, 1978). The availability of small, inexpensive digital processors has resulted in the development of automated techniques for the analysis and acquisition of pulse-height spectra. On-line computer data systems have been successfully incorporated into PHA systems for some time.

The major difference between such systems and a conventional PHA system lies in the ability of the processor to modify information provided by the ADC under control of a programmed sequence of operations that may assume any form the user desires (NCRP, 1978). Figure 6.15 shows a block diagram of a typical processor data acquisition and analysis system suitable for gamma-ray spectra analysis. These computer-based systems are available with a number of complete application software packages. Most of them are able to perform routine counting procedures, peak searches, and identification and quantification of the radionuclides in the sample.

Two basic limitations of all detector-analyzer systems are responsible for most of the difficulties involved in gamma analysis (Lawrence Livermore Laboratories, 1973). First, the finite resolution of any system leads to overlap in peaks of complex spectra. Second, each spectrum invariably includes some background or noise counts (for example, from the Compton spectrum of a higher-energy gamma ray or from external backgrounds).

The simplest procedure for dealing with a complex spectrum is by successively analyzing the various photopeaks beginning with the one with the highest energy. Successive "stripping" subtractions are made. These subtractions remove the Compton portion of that energy from the lower energy area of the spectrum (LLL, 1973). Unfortunately, error analysis is extremely difficult after even a few subtraction steps, so that with complicated spectra the

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uncertainties become large and very difficult to estimate for the lower energy peaks. For this reason, simple "stripping" is used only for the least complicated spectra.

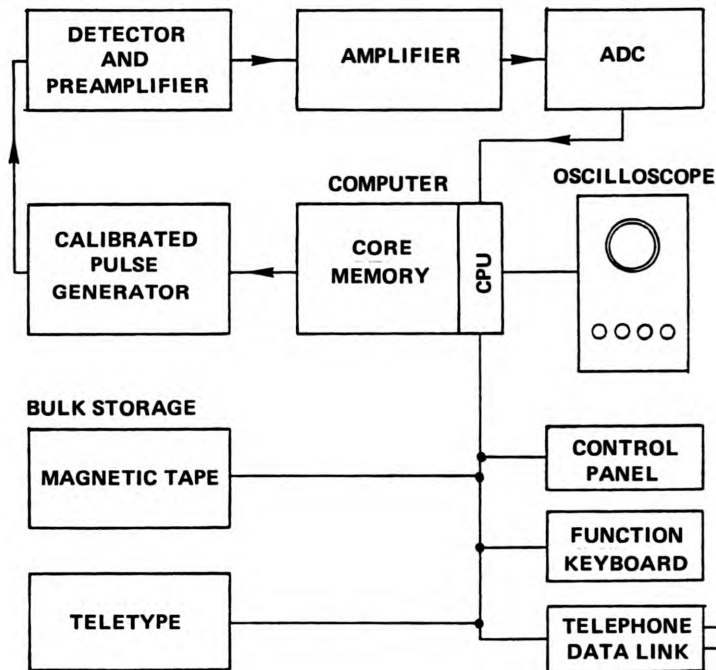


FIGURE 6.15 TYPICAL COMPUTER-COUPLED PULSE-HEIGHT SPECTROMETER.

A somewhat more powerful method is to generate a set of simultaneous equations for a number of radionuclides in terms of their observed photopeak count totals. This technique is somewhat more complex and, again, error problems become very important for the smaller photopeaks or those at lower energies above large backgrounds (NCRP, 1958).

Both of the above are unsophisticated methods, typically using only the full-energy peak information and estimating backgrounds beneath the peaks by measurements in adjacent channels. More elaborate analysis techniques attempt to fit an entire spectrum including Compton-scattering contributions.

One of the most widely used methods of computer analysis is the method of least squares. Essentially, this method uses as input a set of standard spectra taken with the instrument in question and containing known activities of all radionuclides suspected of being present in the unknown sample. A least square fit is then performed on the data; the variances are minimized and the most likely concentrations of each sought-for radionuclide are determined. Among the disadvantages of this method are the requirements that all

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sought-after radionuclides must have reference input spectra available, and that those thought to be present must be "guessed" before analysis.

Multiple regression is an analysis technique which combines the concept of stripping and least squares. Essentially, the method begins with a least-squares fit to the most prominent, highest-energy photopeaks and their accompanying full spectra. The less prominent and lower energy radionuclides are then introduced sequentially, with recomputation of "least squares" each time to improve the goodness of the fit. Each successive recomputation can even be programmed to eliminate a radionuclide previously included. Again, a library of standard spectra must be used (Lawrence Livermore Laboratory, 1973).

6.4 SUMMARY

The detection of ionizing radiation by scintillation counting of visible light is one of the oldest methods on record. This technique of radiation measurement is one of the most useful methods available for the detection and spectroscopy of a wide variety of radiations (Knoll, 1979).

Since this chapter describes a number of detectors utilizing the same basic principle but with a wide range of applications, the summary must be broken into three categories: survey instruments, internal sample detectors, and external sample detectors.

The main advantage of a scintillation type survey instrument are high counting efficiency and the capability of distinguishing between emissions of differing energy. These survey meters are calibrated in the same manner as other survey-type meters.

Internal sample detector systems are particularly applicable to the detection of beta-emitting nuclides. They have a great advantage for the counting of low-energy beta emitters and are limited mainly by their background for low-level counting of more energetic beta emitters. Since the response of the system is proportional to the energy deposited in the detector, counting of three or four radionuclides of differing energy emissions simultaneously is possible. Most commercially available systems are now completely automated so that the efficiency calibration system is built in and efficiency calibration may be performed in a straightforward manner for any type sample.

External sample detector systems encompass most of the good features of detectors and spectrometry systems. Systems employing inorganic materials such as NaI(Tl) as a detector and the computer-based MCA provide a highly sophisticated method for both identification and quantification of radioactivity. The software associated with the computer quickly performs procedures or tasks such as energy resolution determinations, energy calibration, and efficiency calibration.

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CHAPTER 7

SEMICONDUCTOR DETECTORS

J. J. LANZA

7.0 INTRODUCTION

The introduction of semiconductor-type nuclear radiation detectors has significantly advanced the state of the art for the detection of ionizing radiation. They exhibit excellent energy resolution and fast, linear response, which makes them useful in a wide variety of applications. The semiconductor radiation detector can be viewed as a solid-state ionization chamber (Fowler, 1966). In a manner analogous to that of a gas-filled ionization chamber, in a semiconductor detector, charge carrier pairs are formed and swept across the medium by the applied electric field, which gives rise to a detectable electrical current. While in a gas filled ionization chamber the charge carrier pairs are electrons and ions, in a semiconductor detector the corresponding charge carrier pairs are electrons and holes.

A typical semiconductor detector (silicon) requires only 3.6 eV for formation of an electron-hole pair. This compares to 25-35 eV/ion pair for a gas-filled chamber or 50 eV/ion pair for a NaI(Tl) scintillation detector. Because the energy resolution of any detector system is related to the energy required to produce a charge pair in the detector medium, the characteristic low-energy band gap of semiconductor detectors would suggest that they would be superior in energy spectroscopy applications. Typical energy resolution of three of the more common types of detectors for a 500 keV gamma ray would be 3.85 keV full-width-half-maximum (FWHM) for an argon filled ionization chamber, 30 keV FWHM for a NaI(Tl) scintillation detector, and 1 keV FWHM for a germanium semiconductor detector. Thus, semiconductor detectors are clearly superior (and the detector of choice) in high resolution energy spectroscopy measurements.

Germanium and silicon are the semiconductor materials most commonly used as radiation detectors. Other materials that have been investigated include cadmium telluride, cadmium sulfide, and gallium arsenide.

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7.1 PRINCIPLES OF OPERATION

7.1.1 Basic Semiconductor Theory

The excited states of an atom consist of discrete energy levels. An orbital electron can only be found in the ground state or in one of these excited states. The spaces between the excited states are forbidden energy regimes for electrons. In the case of isolated atoms, e.g. a rarefied gas, the states are not infinitesimally sharp but have a certain width, determined by Heisenberg's uncertainty principle. If gas pressure is increased, neighboring atoms begin to influence the width of the states of an individual atom, increasing the width beyond the value demanded by Heisenberg's uncertainty principle. If the particle density approaches solid-state densities, the states become broadened to such an extent that they overlap and form energy bands. Figure 7.1 shows a potential energy diagram for atoms in a crystal lattice, the nucleus being located in a potential well. If neighboring atoms are close enough, as shown in Figure 7.1, e.g., in a crystal lattice, the energy levels are broadened and the potential well between atoms is lowered so that some levels reach over to the neighboring atom. The consequence is that electrons located in these bands no longer belong to a specific atom. They are shared by all atoms. The allowed energy levels have become allowed energy bands. An allowed energy band is a range of energies in which an electron may be found. All other energy bands are forbidden and are called energy gaps or forbidden gaps or bands.

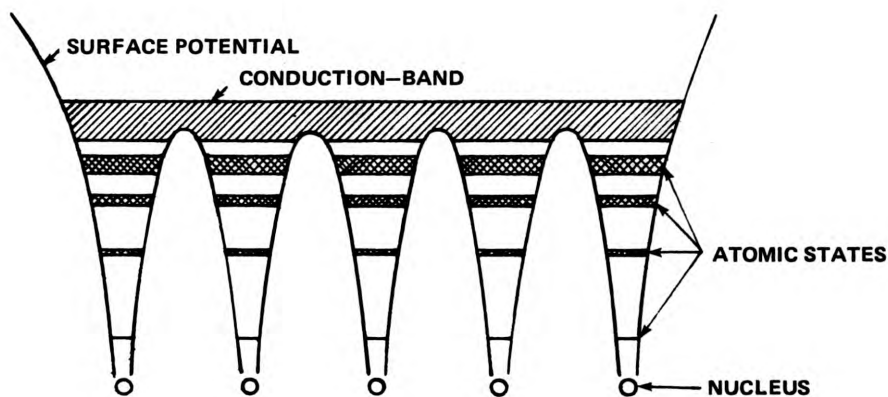


FIGURE 7.1 BAND FORMATION BY CRYSTAL ATOMS.

The energy bands which are important in the operation of semiconductors are the valence band and the conduction band. The conduction band is located at higher energies than the valence band

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and the separation between the two bands is the energy gap (see Figure 7.2).

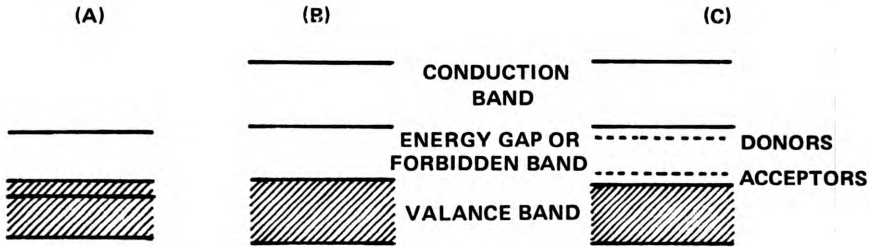


FIGURE 7.2 ILLUSTRATION OF THE IMPORTANT ENERGY BANDS
AND FORBIDDEN ENERGY GAP IN:
(A) A CONDUCTOR; (B) AN INTRINSIC SEMICONDUCTOR;
(C) AN EXTRINSIC SEMICONDUCTOR.

In a conductor, such as graphite or a metal, the conduction and valence bands are contiguous or overlapping, and electrons can move freely from the valence band to the conduction band and will be directed as a current through the conductor upon the application of an electric field (see Figure 7.2a). Differences in the energetics of the interactions of the conduction electrons with the crystal lattice in different conductors account for differences in resistance.

In a substance in which the energy gap between the valence and conduction bands is equal to E (see Figure 7.2b), electrons will be distributed between the two bands according to the Boltzmann distribution law, namely

$$n_c / n_v = e^{-E/kT}$$

where n_c / n_v is the relative distribution of electrons between the conduction and valence levels, respectively, separated by a potential energy E , k is the Boltzmann constant and T is the absolute temperature. From this relationship it can be seen that the proportion of electrons in the conduction band will decrease as E is increased and increase with increasing temperatures.

In a semiconductor the energy gap E between the valence and conduction bands may be of the order of 1 eV and, at room temperature, is of the order of 2×10^{-17} . As the temperature decreases this ratio becomes even smaller, and in a pure semiconductor at absolute zero all valence electrons would be confined to the valence band and a perfect semiconductor crystal becomes a perfect insulator. By contrast, the resistance of a metal generally decreases with decreasing temperatures.

The preceding assumes the semiconductor material contains an equal number of free holes and electrons throughout its volume. Such semiconductors are known as intrinsic semiconductors. Even the

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purest semiconductor crystals are , however, not free from small traces of impurities or of defects in their structure that produce an imbalance in the relative numbers of electrons and holes in the crystal. Such impurities and defects create absorption and emission energy levels, and "electron traps" intermediate between the energy levels of the valence and conduction bands. This presence greatly modifies the luminescent and electrical properties of the crystal.

Most electron traps are situated at energy levels a few hundredths of an electron-volt or more below the conduction band. Electrons that have been excited into the conduction band, in the absence of an applied electric field, will generally migrate around the crystal and some will de-excite to the valence band, emitting radiation while others will interact with a trapping defect or impurity, losing energy and being trapped at an energy level from which transitions to the valence band are forbidden. If the energies of the trapping levels are not too far below that of the conduction band, trapped electrons can again be raised to the conduction band by thermal excitation from the surrounding crystal lattice, thence returning to the valence band with the emission of delayed luminescence. Trapping levels at larger energy differences below the conduction band can only be emptied to the conduction band with the application of higher temperatures, and thermoluminescent dosimeters (TLD's) are based on this principle.

Other types of impurities are those that can either accept an electron creating a hole in the valence band, known as p-type or acceptor impurities, and those that can donate an electron to the conduction band, known as n-type or donor impurities. These impurities are usually situated at energy levels a few hundredths of an electron-volt or more below the conduction band or above the valence band, respectively, for donor and acceptor impurities (see Figure 7.2c).

P-type impurity atoms have only three valence electrons, thus leaving a vacancy in one of the four covalent bands in the semiconductor crystal. This makes it easy for an electron in one of the filled covalent bands between two neighboring semiconductor atoms to fill this vacant band, leaving a hole behind it. In this way, the addition of p-type impurities leads to the formation of excess holes in a semiconductor. Commonly used p-type impurity elements are boron, gallium, and indium.

An n-type impurity atom has five valence electrons leaving an excess electron in the formation of the four covalent bands in the semiconductor crystal lattice. This fifth electron is easily removed, thus becoming a free electron in the conduction band. N-type impurities produce an excess of free electrons in semiconductor materials. Commonly used n-type impurity elements are antimony, phosphorus, and arsenic.

Many applications of semiconductors involve semiconductor material to which either n-type or p-type impurities have been added to produce either, respectively, predominate electron or hole conduction. The intentional adding of impurities is generally known

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as doping. And such doped semiconductor materials are then referred to as an extrinsic semiconductor.

In an insulator the value of E is greater than a few electron volts, which is significantly greater than kT at ambient temperature. This results in correspondingly greater resistivities and reduced conductivity.

For additional information on basic semiconductor theory the interested reader should consult with either Hibberd (1968), Mann, Ayres and Garfinkel (1980), Knoll (1979), Goulding (1978), or Haller (1982) and the references contained therein.

7.1.2 Theory of Operation

The first semiconductor device produced were of the p-n junction type (see Figure 7.3). This means the detector material (silicon or germanium) consists of a p-type material and an n-type material which are joined together creating a space charge region, which is formed at the interface of the two materials by diffusion of electrons and holes. This region is also called the "depletion region" (Durling, 1969).

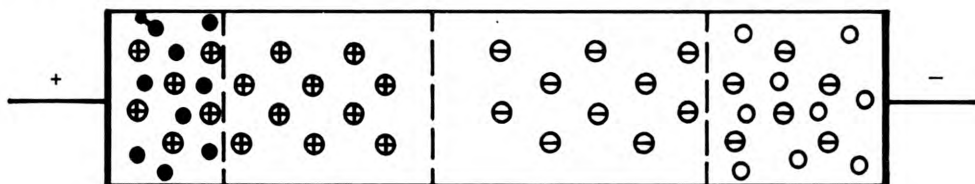


FIGURE 7.3 DIAGRAM OF A p-n JUNCTION SHOWING DISTRIBUTION OF ACCEPTORS ($\ominus$), DONORS ($\oplus$), ELECTRONS ($\bullet$), AND HOLES ($\circ$) IN A REVERSE-BIASED p-n JUNCTION.

Incident radiation entering a semiconductor detector creates free electron-hole pairs by losing energy at a rate of approximately 3 eV per electron-hole pair. If an electric field is present, the electron-hole pair is able to move freely. Incident radiation lifts the electrons in the valence band to the higher energy conduction band leaving holes in the normally full valence bands. This condition exists for about 10 picoseconds after which de-excitation occurs leaving the electrons in the lowest unoccupied band.

The size of the depletion region determines the amount of the useable charge carrier production that will take place in the device. For efficient charge collection, the range of the ionizing radiation should be less than the smallest dimension of the depletion region. The incident radiation must expend all of its energy in the sensitive region if the output pulse height is to be proportional to the energy. The electric field in the sensitive region must be sufficiently large to separate the charge carriers before they can recombine. In general, the carrier lifetime should be several orders of magnitude

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greater than the collection time (Coleman, 1966). The depletion region thickness also determines random noise generation. The junction capacitance is inversely proportional to the depletion width. Therefore, to keep noise as low as possible, the width of the space charge region should be as large as possible (Jones, 1962).

7.1.3 Diode Operational Modes

The photodiode or, more simply, diode configuration is the most commonly used form of semiconductor detector used for the detection and measurement of ionizing radiation. Either the charge impulse, current or voltage output from the detector may be used as a measure of the energy deposited within the sensitive volume of the detector by one or more ionizing events.

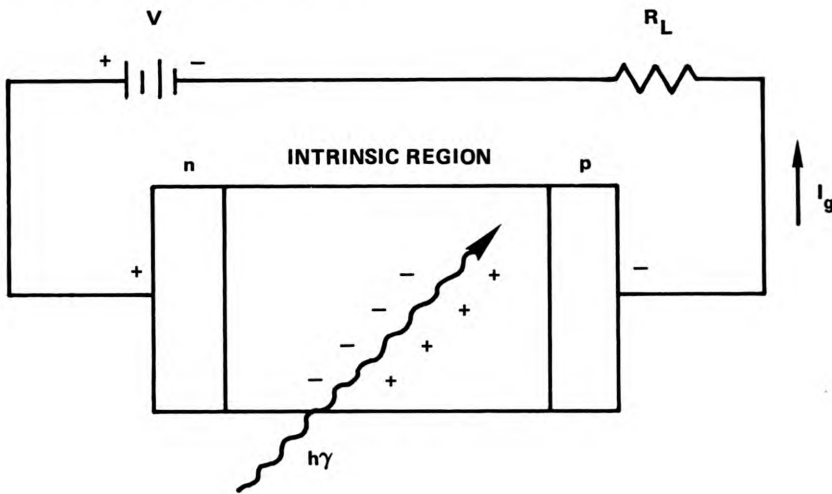


FIGURE 7.4 MODEL OF ELECTRON-HOLE PRODUCTION FROM PHOTON INTERACTIONS IN PHOTOCURRENT MODE OF OPERATION OF A P-I-N JUNCTION DETECTOR (V =EXTERNAL APPLIED BIAS, R_L =LOAD RESISTANCE, I_g =PHOTOCURRENT GENERATED).

The operation of a p-n junction detector requires the application of a reverse bias voltage, as indicated in Figures 7.3 and 7.4. As a consequence, an intense electric field develops across the detector creating a space charge or depletion region devoid of essentially all mobile charge carriers (electrons and holes). This effectively blocks the flow of current through the detector that would occur in the absence of the reverse biased p-n junction. If this were not done, then the current flow through the semiconductor, known as leakage current, would be several orders of magnitude larger than the small current pulses produced by individual ionizing radiation events. Thus, this continuous leakage current flow would completely obscure the observation of the desired events. If an ionizing particle or photon passes through the sensitive region of

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the reversed bias diode, as shown in Figure 7.4, electron-hole pairs are formed. The electrons are swept toward the positive electrode and the holes toward the negative electrode producing a charge pulse proportional to the energy deposited within the sensitive region of the detector. This mode of operation is frequently referred to as the photocurrent mode of operation.

Generally, semiconductor detectors operating in the photocurrent mode are used for spectroscopy. This mode can also be used for dosimetry but it must be realized that the size of the radiation-induced current or voltage signals are determined by characteristics of the individual detector (Jones, 1962). Other parameters that may affect the signal include bias voltage, detector temperature, and load resistance. Therefore, this mode of operation should only be used to measure relative changes in the exposure rate assuming the device is operated in the linear region of the detector response.

Photovoltaic operation involves no externally applied bias; therefore, the width of the depletion region remains constant. One of the best examples of this mode of operation is the silicon solar cell. Under equilibrium conditions, diffusion of charge carriers generates an electric field across the depletion region (Scharf, 1967). The field is directed from the n- to the p-type layers. After irradiation of the depletion region, the electrons are swept towards the n-type side while the holes move toward the p-type layer (see Figure 7.5). Thus, a photocurrent, I_g , is produced consisting of the photovoltaic output current, I_L , in the reverse direction through the external circuit, and a junction current, I_j , in the forward direction. For $R_L = 0$, the so called short circuit DC current, I_L , in the external circuit will attain its maximum value equal to I_g (Scharf and Sparrow, 1964).

The current, I_g , is linearly proportional to the exposure rate, when operated in the short circuit mode. Exposure rates as low as 1.0 mR/min can be measured using semiconductor detectors (Baily and Kramer, 1964).

The photovoltage generated during irradiation (observed as the potential difference across the load resistor produced by the charge output current) can also be used to characterize the exposure rate. However, the photovoltage depends logarithmically on the exposure rate. For this reason, the usefulness of the photovoltage mode of operation is extremely limited as a detection method.

Photocurrent sensitivity to changes in exposure rate and temperature is greatly affected by the input impedance of the measuring circuit. The fact that the junction resistance in semiconductor detectors is temperature dependent regulates the magnitude of thermally generated currents. Therefore, variations in the ambient temperature will not affect the radiation-induced current unless the rate of electron-hole recombination is significantly altered. Thermally generated currents are of concern in the photovoltaic mode only. In photodiode operation, the photocurrents produced are many orders of magnitude higher than these

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so-called "dark currents" for a properly constructed detector (Jones (1963), Scharf and Sparrow (1964), Scharf and Mohr (1971), Lanza (1979), Petushkov and Parker (1973), Klevenhagen (1973, 1977), Bailly and Kramer (1964)).

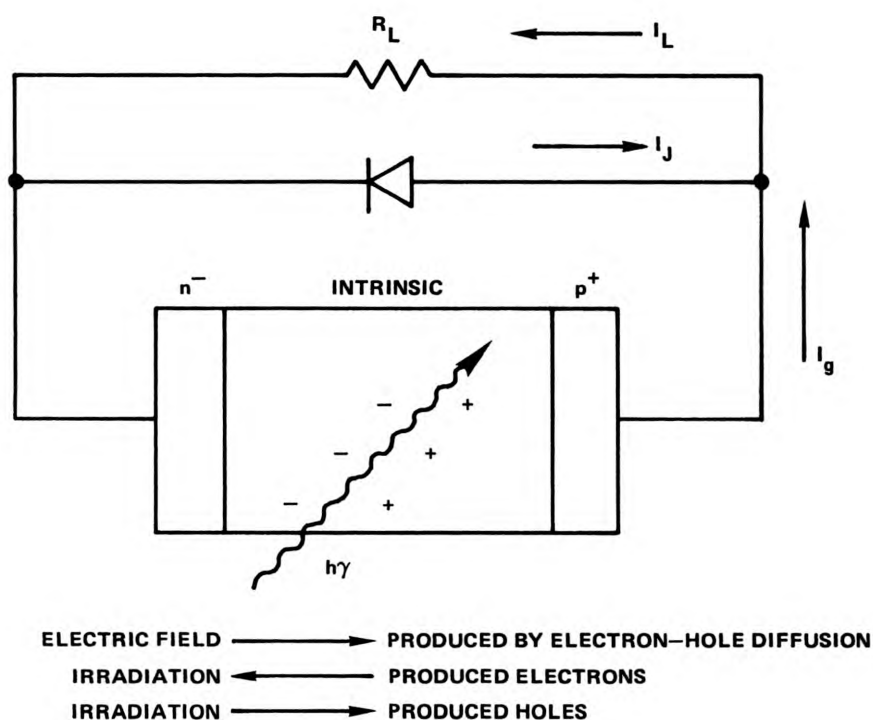


FIGURE 7.5 MODEL OF ELECTRON-HOLE MOVEMENT IN PHOTOVOLTAIC MODE OF OPERATION (R_L = LOAD RESISTANCE, I_L = LOAD CURRENT, I_G = GENERATED PHOTOCURRENT, I_J = JUNCTION CURRENT).

At very low exposure rates (on the order of 10 microR/hr) most semiconductor detectors may be used as pulse counters. In this case, the current generated by the passage of an ionizing radiation gas particle appears as a pulse quite similar to the operation of a proportional counter but not with the associated amplification (except in avalanche detectors). Detection sensitivities similar to that of a GM counter have been reported. However, for moderate to high exposure rates the high sensitivity (efficiencies) of semiconductor detectors produce count rates that exceed the capabilities of most conventional pulse counting techniques (Fowler, 1966).

The modes of detector operation previously discussed were concerned with transitory changes in the electric characteristics of the semiconducting material. However, the conductivity of a detector can be altered by damage to the structure of the detector material caused by ionizing radiation. This change in electrical conductivity can also be used to determine absorbed dose (Fowler,

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1966). Such measurement techniques are seldom, if ever, used as they produce irreversible damage to most semiconductor detectors.

7.1.4 Energy Resolution

During the interaction of ionizing radiation with a semiconducting material, energy, E , is deposited in the material resulting in the the formation of N electron-hole pairs according to the relationship:

$$N = \frac{E}{W} \quad (7.1)$$

where W is the energy required to produce one charge carrier pair. W is smaller in semiconductors than in scintillation and gas ionization systems; the resultant current signal from an ionizing interaction is, therefore, larger. Consequently, statistical spread in the number of charges produced is smaller (Palms (1971), Mann, Ayres, and Garfinkel (1980)).

The energy resolution of a system defines its ability to resolve the energy difference between two monoenergetic radiation sources. The smaller the difference that can be resolved, the higher the resolution of the system. The resolution of a semiconductor detector spectroscopy system is determined by the following four parameters:

1. Statistical variations in the number of electron-hole pairs created.
2. Noise in the detector caused by leakage and other sources.
3. Electronic noise in the amplifying system.
4. The time stability of the electronic detecting system (NCRP 58, 1978).

The statistical variation in the number of electron-hole pairs generated is given by the limit for the best resolution of the system.

Random noise, found in the detector and the associated electronics, will broaden the obtained peak produced by monoenergetic particles. Much of this noise is temperature dependent and may be reduced by cooling the detector and the preamplifier input electronics.

Although the introduction of well designed solid state amplifying systems has greatly reduced drift problems, constant attention to system operational characteristics must be observed. Amplifiers featuring field-effect transistor (FET) input stages have been used for many years and have vastly improved the sensitivity of detection systems. However, decrease of energy resolution can occur if foreign particles or moisture is allowed to accumulate in connectors and other electronic components.

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Figure 7.6 shows a comparison of a spectrum obtained with a 3 x 3 inch cylindrical NaI(Tl) detector (discussed in Chapter 6) and one obtained with a typical semiconductor detector. It can be seen that the energy resolution of the semiconductor detector is vastly superior to that of the scintillation detector.

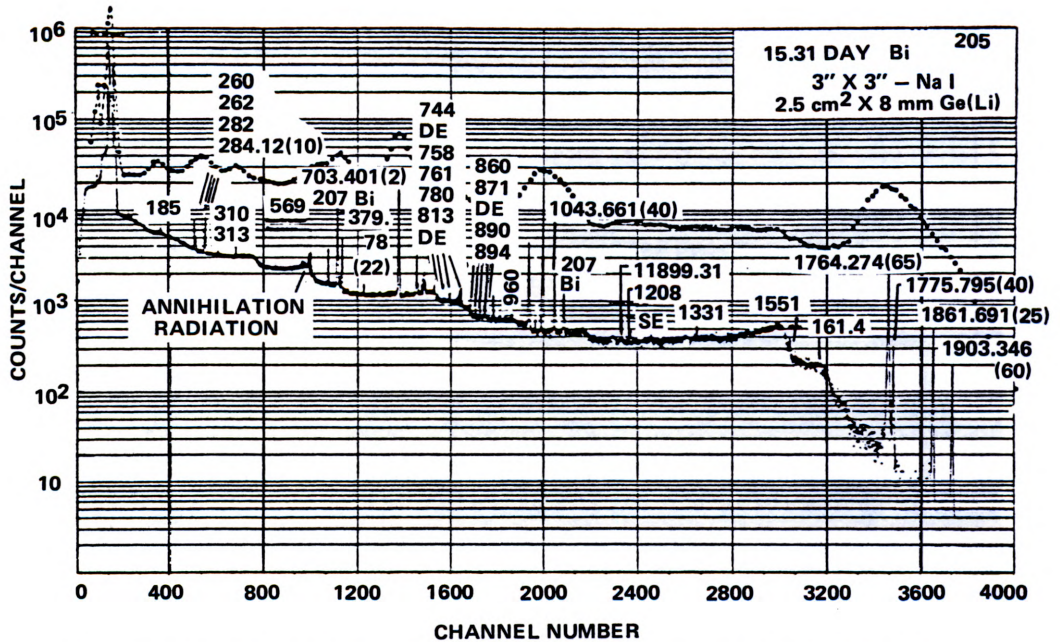


FIGURE 7.6 PULSE-HEIGHT SPECTRUM OF GAMMA RAYS EMITTED IN THE DECAY OF ^{205}Bi OBTAINED WITH A Ge(Li) DETECTOR. THE RESPONSE OF A 3" X 3" CYLINDRICAL NaI(Tl) DETECTOR IS SHOWN TO ILLUSTRATE THE IMPROVED ENERGY RESOLUTION OF SEMICONDUCTOR DETECTORS. "SE" AND "DE" INDICATE SINGLE- AND DOUBLE-ESCAPE PEAKS (NCRP-58).

7.1.5 Efficiency

As discussed in Chapter 6, size and composition of the detector greatly influences the system efficiency. The higher the atomic number (Z) of the material, the greater its stopping power for ionizing radiation. Thus, a germanium semiconductor with a Z of 32 will have a greater efficiency than silicon with a Z of 14 for highly penetrating radiation, such as, high energy photons. Also, the larger the volume of the detection medium, the greater are the chances for interaction with an external source of radiation. This applies to primary interactions as well as secondary ones arising from Compton-scattered photons.

The source-detector geometry can be varied in a number of ways to increase efficiency. The physical shape and the dimensions of the detector may be varied to improve the acceptance angle for the

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radiation flux. The position of the source relative to the detector is also an important factor determining overall detector efficiency. The use of several detectors positioned around the source in a way to capture most of the emitted energy may also be employed.

The efficiency of semiconductor detectors exhibits a strong energy dependence. Figure 7.7 shows the efficiencies for different volume Ge(Li) detectors as a function on energy. Often the response of a 3 x 3 inch NaI crystal is used to compare the efficiencies of semiconductor detectors.

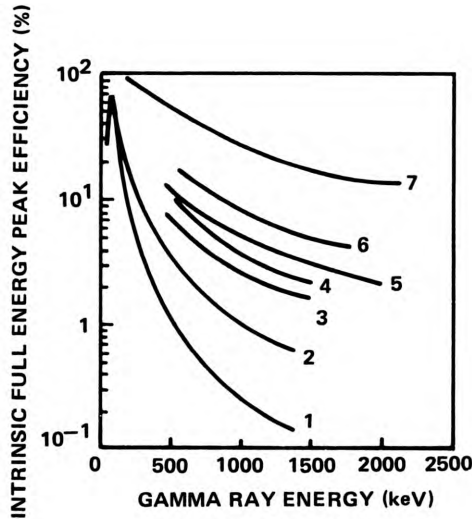


FIGURE 7.7 EXPERIMENTAL FULL ENERGY PEAK EFFICIENCY FOR DETECTOR OF DIFFERENT VOLUMES. CURVES 1 AND 2 REFER TO PLANAR DETECTORS, WHILE CURVES 3, 4, 5, AND 6 REFER TO COAXIAL DETECTORS. CURVE 7 IS FOR A 3' X 3' NaI(Tl) DETECTOR.

7.1.6 Semiconductor Detector Types

Semiconductor detectors may be used for counting samples containing alpha, beta or gamma ray emitters and can be divided into several categories according to design features and operational characteristics. Two main types presently available are: the surface-barrier silicon p-n junction detectors, which are mainly used to detect alpha-particles and beta-particles; and, the lithium-drifted silicon or germanium p-i-n (which may be constructed from either lithium-drifted or hyper pure Ge) junction detectors, which are predominantly used for photon spectroscopy.

Junction Detectors

There are two types of p-n junction detectors presently in

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common usage: surface barrier and diffused junction.

Silicon surface barrier detectors with partially or totally depleted sensitive regions are used mainly for charged particle spectroscopy. Diffused junction detectors can be used for spectroscopy as well as dosimetry involving charged particles or photons.

The typical silicon surface barrier detector is a large area diode made by allowing an oxide layer to form on the surface of a piece of high resistivity (n-type or p-type) silicon material. Electrical contacts are made to the p- and n-type surfaces. An externally applied reverse bias creates an electrical field across the depletion region. The depth of the sensitive region should be chosen to match the maximum energy of the radiation of interest. For maximum collection, totally depleted detector material sections are coaxially stacked to obtain a resulting depth of material that will absorb the total energy of the incident radiation.

Since the bandgap energy is fairly large in silicon, interband transition caused by thermal energy is low at room temperature. Therefore, surface barrier detectors can be operated at low noise levels at room temperatures without the need of troublesome and complicated cooling systems.

Totally depleted devices are ideal for counting of beta particles. Compared to ionization chambers, silicon detectors have better resolution and show better stability. They can be operated essentially windowless. This latter characteristic of the detector allows it to detect alphas, fission fragments, protons, and other charged particles. Silicon detectors have close to 100% efficiency provided the energy of the charged particle deposited in the sensitive region of the detector is much larger than the noise level in the device. Backscatter losses are the only significant degrading effects. For counting purposes, the width of the depletion region need only be large enough to assure that the energy loss of the particle is sufficient to trigger the discriminator of the system (Price, 1964).

Totally depleted silicon detectors provide good resolution when used as spectrometers for beta, proton, alpha particles and fission fragments. For low energies, a single detector can be used to collect all the energy of the particle in its sensitive region, a characteristic necessary for spectroscopy. Sensitive depths in surface barrier detectors are available from 10 to 2000 micrometers.

Although not designed to count gamma rays (due to the low Z of silicon), totally depleted silicon detectors can be used in place of the more expensive and less convenient germanium detectors for relatively efficient detection of low-energy gamma photons.

Diffused junction detectors are partially depleted devices. They are fabricated from high resistivity p-type silicon into which phosphorous has been diffused to form the p-n junction. The resulting device is more rugged than the surface barrier detector but the need for high temperature (800°C) in its manufacture often causes degradation in the physical characteristics of the material unless

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they are fabricated by ion implantation techniques (Harshaw, 1978).

Diffused junction devices have thin windows (electrical contacts). The sensitive depths (up to 100 micrometer) depend on the resistivity of the base material and on the applied potential difference. They may be used for alpha spectroscopy and gross charged particle counting applications.

A more useful junction device is the doubly diffused silicon detector, commonly called a p-i-n diode, in which the 'i' symbolizes the greatly increased intrinsic region width. In manufacturing this double-junction detector, boron (p+) is first diffused into a high resistivity (2000 ohm-cm) silicon wafer. The "+" designation signifies a heavily doped junction. Then, phosphorous (n+) is diffused into the p+ substrate. The width of the intrinsic region, i, is determined by the depth of the phosphorous diffusion and by the width of the wafer. Sensitive depths up to several millimeters have been achieved (Blankenship and Borkowski (1962), Ziemba et al. (1962), Ramondetta and Groeber (1978)).

The silicon p-i-n junction has begun to gain considerable popularity as a beta, gamma, and x-ray dosimeter. When used in the photocurrent, photovoltaic, or pulse counting mode, the p-i-n junction devices have been shown capable of dose measurements from 10^{-6} rad/min to 10^6 rad/sec (Urie (1979), Baily and Hilbert (1965)).

Lithium-Drifted Devices

In the early stages of development, it was not possible to achieve high purity semiconductor materials (germanium or silicon) to allow complete collection of charges at depths of 1 or 2 cm. Impurities in these materials provided trapping sites which prevented total charge collection. To overcome this problem, lithium (Li^+) ions were drifted under a bias into the semiconductor substrate to compensate for the p-type impurities. This allowed large detectors, having a volume as large as 100 cm^3 , to be developed. Consequently, germanium-lithium drifted (Ge(Li)) and silicon-lithium drifted (Si(Li)) detectors developed into the most popular devices for x- and gamma-ray spectroscopy.

Ge(Li) detectors are primarily used for gamma spectroscopy in energies ranging in excess of 100 keV. These detectors possess good resolution (ΔE of several keV as shown in Figure 7.5), and high efficiencies (up to 75% of NaI), but must be kept and operated at liquid nitrogen temperatures due to the high mobility of the lithium-ions in germanium at room temperatures.

These detectors are now available in various geometries. Planar p-i-n diodes were the earliest shapes developed. They consist of an intrinsic (compensated) region on a p-type substrate and are covered by an n-type surface through which the photons enter. However, the practical limit of lithium drift (15-20 mm) delineates the useful volume that can be achieved. In order to provide a larger intrinsic region, the coaxial p-i-n detector was developed. High

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quality p-type germanium is used. The compensated region is produced by lithium drifting while the central core remains p-type material. The outer surface is overcompensated by the presence of lithium and forms an n-type region. Figure 7.8 shows two kinds of coaxial detectors - the true open-ended and the wrap-around closed-end coaxial.

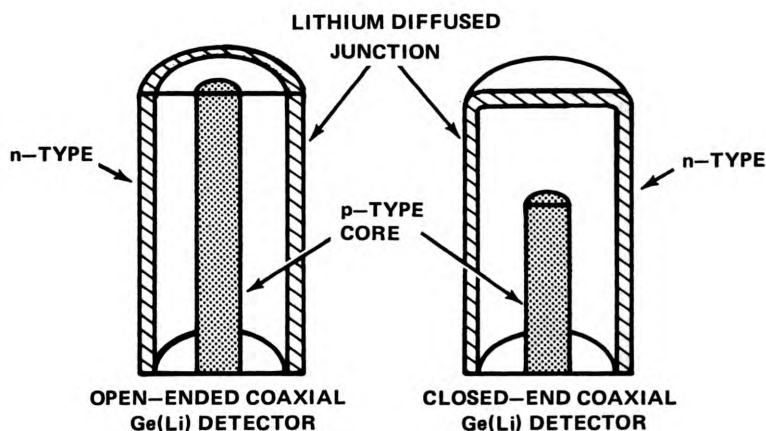


FIGURE 7.8 SCHEMATICS OF CYLINDRICAL ANNULAR AND SEMI-ANNULAR TYPES OF Ge(Li) DETECTOR (HEATH, 1969).

A major advantage of the coaxial geometry is that large volumes can be compensated. Also, the electric field increases toward the axis resulting in faster drift velocities than achieved in a planar detector, which has a uniform field. This in turn yields better energy resolution by reducing the probability that an electron will be trapped near the core rather than making its way to the core where it will contribute to the collected charge. Furthermore, leakage currents are low since the exposed junction area at the end of the cylindrical detector can be kept relatively small, especially for long cylinders, resulting in large sensitive volumes. The primary disadvantage of very long cylinders is that the capacitances are larger.

Ge(Li) detectors, because of their good energy resolution, are used extensively in spectroscopy systems for a wide variety of high energy photon applications. Experience has demonstrated that the superior resolution of Ge(Li) detectors compared to NaI(Tl) detectors is often advantageous in many applications.

Silicon is not widely used for gamma spectroscopy except at low energies. However, because of the good response at energies below 100 keV, Si(Li) detectors are used extensively for low energy photon spectroscopy, especially x-ray detection.

One key advantage of Si(Li) is that it can be operated at room temperatures for many applications. This eliminates the need for windows and makes spectrometry of very low energy photons (a few keV) much easier. For best resolution, however, low temperatures are

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still necessary.

One disadvantage is the lower density of silicon compared to germanium. Another is that its intrinsic resolution is not quite as good, although when operated close to 0°C the differences are slight.

Small size silicon detectors are often used when it is necessary to count low-energy photons in the presence of a large background of high energy photons since the detection efficiency for the higher energies will be quite small.

Intrinsic Detectors

Single crystals of germanium can now be produced with sufficient purity for fabrication of high-quality detectors that no longer require lithium drifting to compensate for impurities. Large volume planar as well as coaxial detectors having volumes approaching those of comparable lithium drifted devices are now available and such detectors have several advantages. Since lithium-ion mobility is no longer a problem, these detectors do not have to be maintained at cryogenic temperatures at all times. Although other problems exist, these devices appear to be much more stable and reliable under adverse environments. However, it should be stressed that these devices also need to be operated at liquid nitrogen temperatures to reduce thermally generated electrical noise. These intrinsic or hyper pure germanium detectors are rapidly replacing their corresponding lithium drifted types in popularity.

Avalanche Detectors

A major disadvantage of p-i-n and Li-drifted devices is the absence of any internal gain mechanism, necessitating the use of low-noise and high-gain pulse amplifiers. In more recent developments, the surface of the sensitive region is contoured so that the internal electric field is raised sufficiently for avalanche multiplication of carriers to take place without the onset of surface breakdown. The principle advantages are the sensitivity to x-rays below 1 keV and betas as low as 6 keV, very low background (1 count per minute), and high count rate capability (several million counts per second).

Figure 7.9 shows a silicon avalanche detector employing a gallium-diffused junction. Normally, for optimum charge collection, the device is operated at a reverse bias within 2% of the avalanche breakdown voltage. When radiation strikes the detector, electron-hole pairs are formed initiating an avalanche of 10^2 - 10^3 electrons for each radiation induced electron-hole pair. This electron avalanche is swept across the junction and collected. This concept of charge multiplication is similar to that underlying the operation of the gas proportional counter. Avalanche detector noise levels are low which can be interpreted as being equivalent to an increase in the effective energy deposition (Moldofsky and Gelezunas, 1972).

The low background levels in this device preclude the need for

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large amounts of shielding and anticoincidence electronics for temperatures up to 100°C . Backgrounds of less than 1 cpm are achievable, allowing very low emission rate sources of a few cpm to be measured (Moldofsky and Gelezunas, 1972).

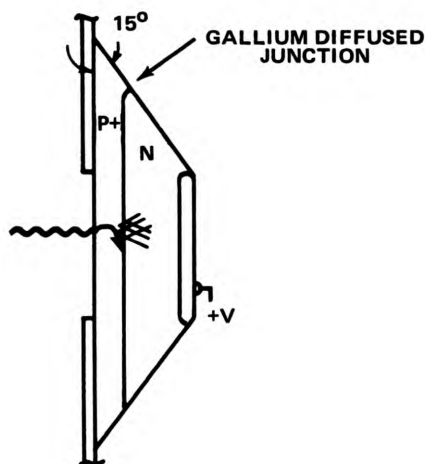


FIGURE 7.9 CROSS-SECTION VIEW OF AN AVALANCHE DETECTOR (MOLDOFSKY, 1972).

The avalanche detector operated as a pulse counter has been proven useful in x-ray, gamma, and beta detectors as well as in high energy (5 MeV) neutron detectors.

Bulk Conductivity Detectors

This category of semiconductor materials includes those made of cadmium sulfide, cadmium telluride, and gallium arsenide (Entine (1979), Urie (1979), Luthmann (1979), Allison (1978)). CdTe is discussed in this section since it is the most commonly used.

CdTe is a high Z ($\text{Cd}=48$ and $\text{Te}=52$) material with an energy bandgap of 1.5 eV at room temperature. This relatively large bandgap allows operation at room temperature without excessive leakage currents. The mechanism of current conduction in these detectors is no longer analogous to that of a gas-filled ionization chamber. Bulk conductivity detectors such as CdTe exhibit a current gain, or amplification mechanism and therefore perform in a manner similar to that of a proportional counter (Fowler, 1966).

Figure 7.10 shows the concept of the current conduction mechanism. An ionizing radiation particle interacts with the detector material and as a result an electron-hole pair is generated. This means an electron is lifted into the conduction band and a hole is left in the valence band. Conduction as a result of an applied field can be accomplished by the electron in the conduction band as well as by the hole in the valence band.

Usually after a short time, the electron-hole pair will

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recombine and conduction will cease (see Figure 7.10 (b)). However, if acceptor levels (electron traps) should be present in the detector material, some of the electrons can fill these levels and be prevented from recombination with holes, thus leaving some holes behind in the valence band. When this occurs conduction still takes place. The addition of the acceptor levels thus has the effect of amplifying the initial charge impulse by permitting a flow of current over a much longer time interval, that is, until the holes eventually recombine with the trapped electrons.

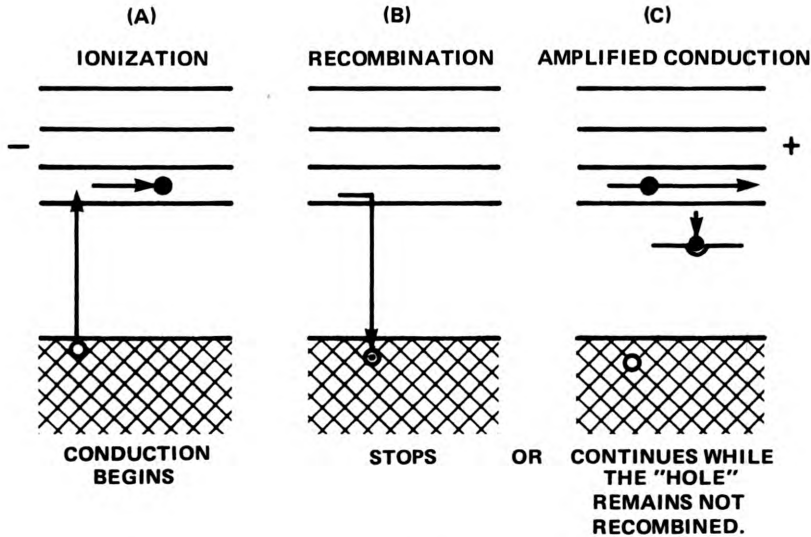


FIGURE 7.10 SCHEMATIC REPRESENTATION OF THE CURRENT INDUCED IN A BULK CONDUCTIVITY DETECTOR BY THE CREATION OF ONE OR MORE CHARGE CARRIER PAIRS (FOWLER, 1966).

CdTe has been used for counting ionizing radiation pulses as well as for spectroscopy. CdTe detectors are efficient counters since the high Z of the detector material stops a large percentage of the incident gamma rays. Sensitivity is also high due to the low background noise which allows for the utilization of a large percentage of the obtained counts. However, the amplification mechanism (trapping of one component of the charge pair) results in long pulse periods. This means such a detector will exhibit a large dead time.

For applications which involve more than one energy of gamma radiation, or where it is important to differentiate between the small differences in the energies of unscattered and scattered radiation, a detector with a better energy resolution than is obtainable by a counter is required. CdTe energy discriminators provide sufficient resolution for many of these applications. The convenience of room temperature operation and small detector size are additional attractive features. However, despite claims made indicating very high resolution, experience has shown this is not the

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case with the present generation of CdTe materials available (Urie (1979), Allison (1978)).

Problems dealing with the energy resolution of CdTe are due to two properties exhibited by the semiconductor material. CdTe shows "tailing" which means a source appears to yield a large number of events with an energy distribution ranging from the true source energy down to noise levels. In addition, the position of the photopeak appears to shift its location in the spectrum as the length of time the bias voltage has been applied increases. This effect has been termed polarization (Chynoweth (1952), Allison (1978), Urie (1979)). These factors limit the usefulness of CdTe in spectroscopy.

Since it can be machined into special shapes, can be operated at high temperatures, and shows good counting efficiencies, the use of CdTe as a pulse counter for the measurement of ionizing radiation is attractive.

7.2 NORMAL OPERATIONAL USE

Semiconductor detectors are mainly used for spectroscopy and for the detection and dosimetry of heavy particles as well as photons. The advantages of semiconductor materials include features such as fast response times, high energy resolution, and good charge collection efficiency. For this reason, this type of detector has replaced the more conventional gas and scintillation detector systems in spectroscopy and dosimetry applications.

7.2.1 Operational Mode - Detection and Dosimetry

Semiconductor detectors provide a simple means of detecting and quantifying emissions from radiation sources. All of the semiconductor detector types previously mentioned (Section 7.1) have been employed for these purposes. Presently, however, avalanche diodes, silicon and germanium junction, and bulk conductivity detectors (CdTe) are predominantly used with various commercial and experimental radiation monitoring systems.

The availability of sophisticated electronic components has greatly influenced the utilization of semiconductors as dosimeters. Sensitive electrometer amplifiers using FET-input transistors have allowed the measurement of extremely small currents when low fluence sources are present.

One of the advantages of semiconductor materials is their ability to be shaped into a variety of forms and sizes depending on the intended applications. In medicine, small silicon junction, avalanche, and CdTe detectors are being used as sensitive beta particle counters when placed into catheters and inserted in vivo (Huth and Moldofsky (1971), Moldofsky and Gelezunas (1972), Palms (1971)). Silicon p-i-n diodes when arranged in a proper geometry

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have shown minimal directional dependence for gamma and x-ray beams of varying energies (Lanza (1979), McKetty (1978)). Avalanche and silicon p-i-n diodes have also been used as integrating neutron dosimeters for personnel (Ramondetta and Groeber, 1978). A new radiation "chirper" which takes advantage of CdTe high sensitivity (and therefore small detector size), high reliability, and low power requirements has recently been introduced for radiation monitoring for personnel. Also, a CdTe wound monitor utilizing its small detector size for highly directional contamination localization has been developed.

Recent technological advances concerning material purity and electronic systems interfacing techniques have had great impact on improvement of semiconductor detectors and their utilization. Nuclear medicine applications have benefited greatly by the introduction of CdTe and intrinsic germanium detector systems (Mauderli (1979), Entine (1979)). The future for semiconductor devices, especially in dosimetry, will be marked by an increasing complexity of systems employing detectors such as avalanche diodes or bulk conductivity devices performing spectroscopic analysis of incident radiation fluxes. Computer assisted evaluation of the energy spectrum is possible with the high resolution energy selective semiconductor detectors thereby permitting the biological effectiveness of the detected radiation to be related to an equivalent absorbed dose.

7.2.2 Operational Mode - Spectroscopy

Semiconductor detectors have been used for the spectroscopy of alpha and beta particles, as well as x- and gamma rays. Presently, perhaps the most important application of semiconductor detectors is their use as gamma and x-ray spectrometers. This is due to the very high energy resolution available, plus often broader energy ranges and greater efficiency than other high resolution instrumentation such as crystal and magnetic spectrometers (Fowler (1966), Goulding and Stone (1971)).

Gamma Spectroscopy

Due to their ability to clearly separate closely spaced gamma peaks, semiconductor detectors, and specifically Ge(Li) detectors have become a vital part of an analytical laboratory. For energies greater than 50 keV, large volume coaxial Ge(Li) or hyper pure Ge detectors provide a sensitive and convenient means of radionuclide analysis at environmental as well as reactor-activated radiation levels. This good sensitivity is due to the large peak-to-Compton continuum ratios that can be obtained with these detectors. The Compton continuum is produced by the passage of gamma rays through the sensitive volume of the detector without the deposition of the full energy of the event and adds to the background events against

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which the actual gamma induced photopeaks must be detected.

Gamma spectroscopy has been used extensively for radionuclide analysis of emissions at the environmental level (i.e., low-level) in discharges from nuclear power plants as well as releases from conventional fossil fuel burning facilities. The use of these techniques has been standardized in the before-operation and after-startup phases of an environmental surveillance program in areas surrounding these activities. In-plant monitoring of reactor primary and secondary coolant systems as well as discharged elements is routinely performed using high resolution semiconductor detector systems.

The field of neutron activation analysis (NAA) has been greatly advanced by the use of high resolution Ge(Li) detectors. NAA has been applied to a wide variety of investigations including forensic analysis of evidentiary materials (especially hair samples), identification of toxic heavy metals in biological samples, and provenience studies on rock and pottery samples. Neutron activation analysis requires a high fluence source of neutrons such as can be provided by a nuclear reactor or accelerator, in order to achieve activation levels sufficient for good statistical analysis by the detection system.

X Ray Spectroscopy

The determination of the elemental concentration in a sample by means of characteristic x rays is a technique now made very convenient through the use of Si(Li) and low photon energy Ge(Li) detectors. The method involves irradiating the sample with x rays or heavy charged particles to induce x-ray fluorescence. This means electrons are removed from the inner atomic shells of the atoms in the sample by the irradiation. The created vacancies are filled by electrons of the higher lying shells. The resulting transitions produce the characteristic x-ray lines which are particular to each element. Previously, optical spectroscopy (emission and atomic absorption) have been extensively used (Goulding and Stone, 1971). However, x-ray fluorescence is simpler and, as it turns out, in most cases more sensitive.

This technique is used in combination with scanning electron microscopes and electron microprobes for samples as small as 1 μm in diameter. Using protons or other heavy charged particles, samples weighing less than a nanogram can be accurately analyzed.

7.2.3 Detection Systems

Detection systems for radiation measurements have evolved from the classic gold-leaf electroscope which measured the accumulated charge produced by the passage of ionizing radiation through the air surrounding the device. In the meanwhile, electronic systems have been developed to adequately detect the very small charges induced by

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the passage of ionizing radiation through semiconductor materials.

Most of the following discussion directly describes detector systems designed for a Ge(Li) crystal used for gamma or x-ray detection and spectroscopy. Similar systems can be used with suitable modifications for other types of semiconductor detectors.

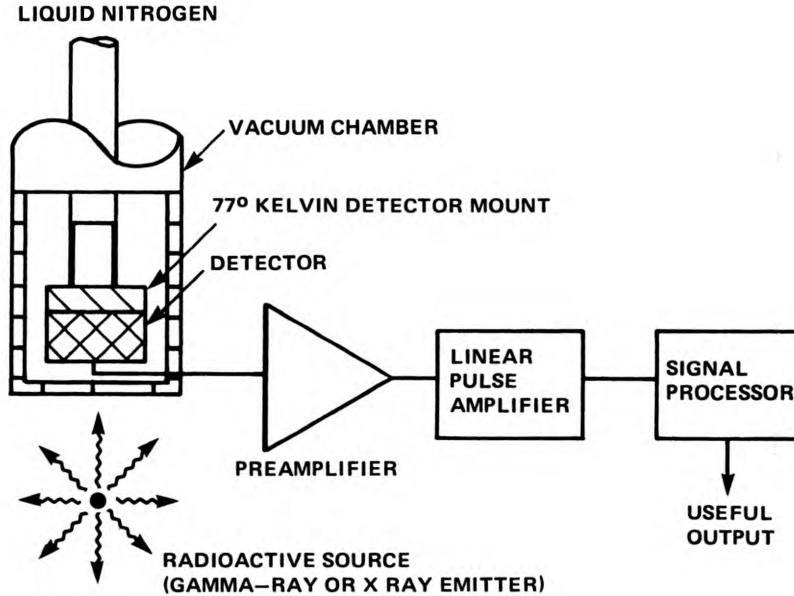


FIGURE 7.11 BASIC COMPONENTS OF SOLID-STATE LITHIUM-DRIFTED GERMANIUM DETECTOR SYSTEM (BRADLEY, 1971).

The basic components for a Ge(Li) type gamma ray analysis system as shown in Figure 7.11 are (Bradley, 1971):

1. The Ge(Li) radiation detector (with cooling system).
2. A preamplifier that provides low-noise amplification of the detector output.
3. A main amplifier providing further amplification, and pulse shaping to maximize the signal-noise ratio.
4. An analysis system to sort and display the energy distribution of the detected events.

The analysis system may involve a single channel analyzer (SCA) which will pass only those pulses falling within a preselected range of amplitudes that correspond to a particular energy region, or it may be a multichannel analyzer (MCA) that performs pulse-height analysis of pulses by sorting them into individual bins or channels that correspond to the energies of the events producing the pulses.

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7.3 CALIBRATION

The calibration of semiconductor detectors is similar to procedures performed on other radiation measuring devices with only a few variations. For dosimetry and counting purposes, a direct comparison between the measurement values obtained for the source of unknown activity with that of a previously calibrated source is generally used. However, when spectroscopy is to be performed, as described in Chapter 6, calibration must include studies of: 1) energy calibration; 2) resolution determination versus energy; and, 3) efficiency versus energy. The geometry of the detector with respect to the source is vitally important with semiconductors because of their construction and because of the uncertainty as to the exact location and depth of the sensitive region under a particular operating condition.

7.3.1 Factors Affecting Calibration of Detectors

The few basic factors that affect the ability of a semiconductor to detect the presence of radiation are described in Chapter 6. In this section and in Section 7.3.2 the calibration of semiconductor detector systems for use in counting, dosimetry, and spectroscopy roles, respectively, will be discussed.

Most semiconductor detectors can be used to count pulses (event counters) with a minimum of signal processing electronics. In fact, this mode of operation is probably their most sensitive since radiation levels down to $1.0 \mu\text{R/hr}$ have been detected using charge-sensitive preamplifiers.

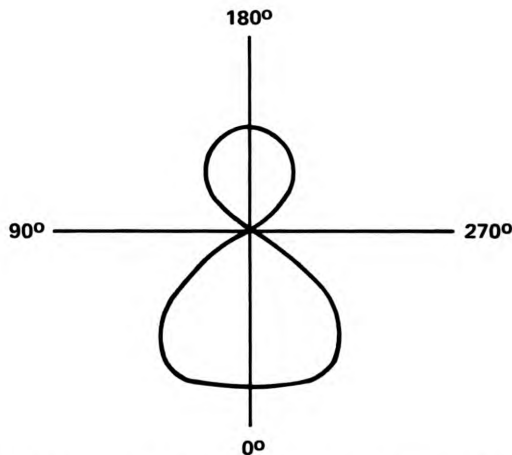


FIGURE 7.12 ANGULAR RESPONSE OF A TYPICAL SEMICONDUCTOR DETECTOR (SHOWING IMPORTANCE OF SOURCE-DETECTOR GEOMETRY).

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For simple event counting applications, the avalanche diode will probably be used most frequently in the future due to its great sensitivity and high intrinsic signal amplification. Figure 7.12 shows a plot of the angular response of an avalanche detector to 17 keV x-rays. From this figure, it can be seen that an accurate knowledge of the source-detector geometry must be known. This is true for all planar geometry detectors. Coaxial detectors can have essentially 360 degree rotational response uniformity under certain conditions. For avalanche diodes (or any planar detector), Figure 7.13 shows a possible configuration geometry that has been shown to be effective in producing a fairly uniform 360 degree signal response (McKetty, 1978). Figure 7.14 shows the efficiency of a typical avalanche detector to x rays and neutrons.

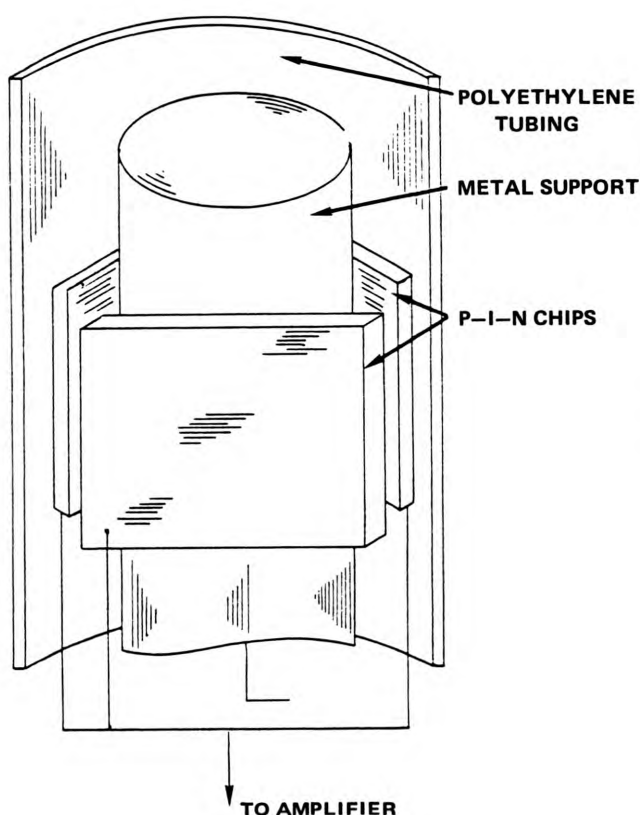


FIGURE 7.13 COMPENSATION FOR DIODE DIRECTIONAL DEPENDENCE USING 120° DETECTOR EMBLACEMENT (McKetty, 1978).

Calibration of the actual diode detector (in this case an avalanche diode) by a source (an x-ray machine) relates exposure-rate (mR/hr) to the radiation flux (rays or particles/cm²-sec.) which in turn produces a specific count rate in the diode. Figure 7.15 shows

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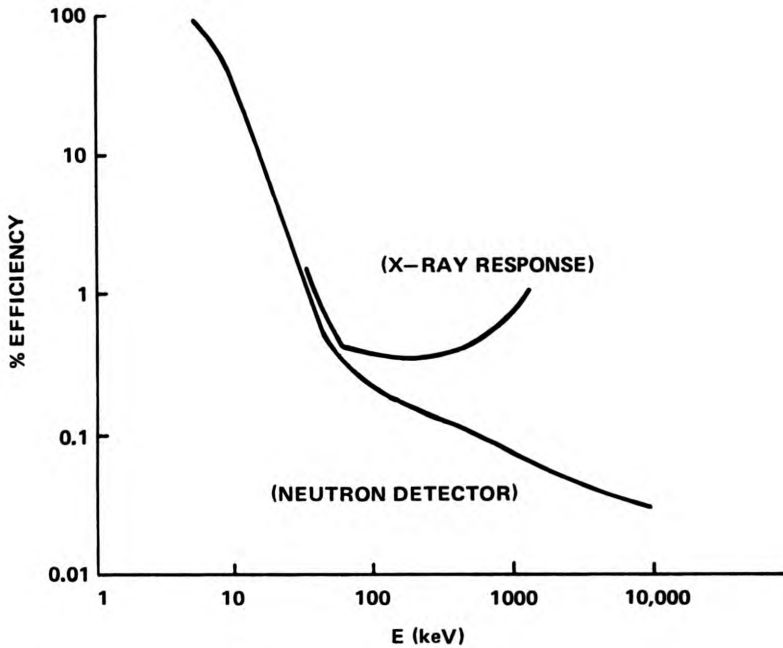


FIGURE 7.14 DETECTION EFFICIENCY AS A FUNCTION OF ENERGY FOR X-RAYS AND NEUTRONS OF A TYPICAL AVALANCHE DETECTOR (AFTER MOLDOFSKY, 1972).

the x-ray flux-density versus energy for a typical x-ray machine corresponding to an exposure rate of 1 R/hr (Moldofsky and Gelezunas, 1972). This curve was obtained using a conventional NBS-calibrated Victoreen R-meter chamber (secondary source) and checking the exposure rate over a range of selected energies. This value was then converted to a photon flux density for the source which when divided into the actual count-rate produces the efficiency curve previously shown in Figure 7.14. Figure 7.16 shows the exposure response of four diodes in counts-per-second versus energy. At this point, a curve of count-rate versus exposure-rate can be generated as depicted in Figure 7.17.

A similar procedure can be used to calibrate other types of diodes in the previously described operational modes: i.e., photovoltaic or photocurrent. Figure 7.18 shows a plot of photocurrent versus exposure rate for silicon p-i-n diode operated in the photovoltaic mode. For diodes operated with an applied external bias, it must be remembered that the calibration is only good for the one voltage, as demonstrated by the effect of varying the bias voltage of a detector as shown in Figure 7.19, at which a particular detector was calibrated.

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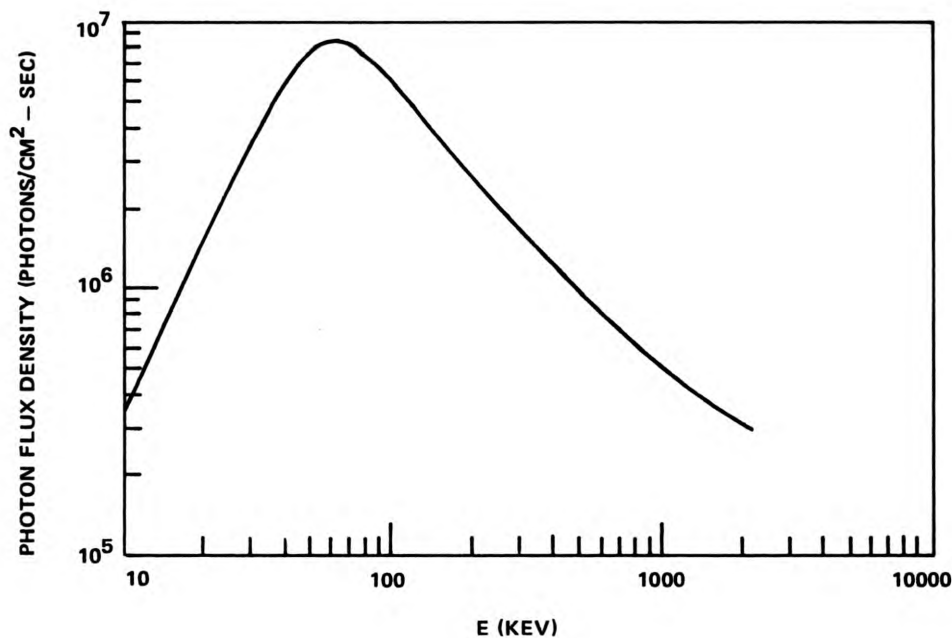


FIGURE 7.15 FLUX CORRESPONDING TO 1 R/hr AS A FUNCTION OF INCIDENT PHOTON ENERGY FOR A TYPICAL X-RAY GENERATOR (MOLDOFSKI, 1972).

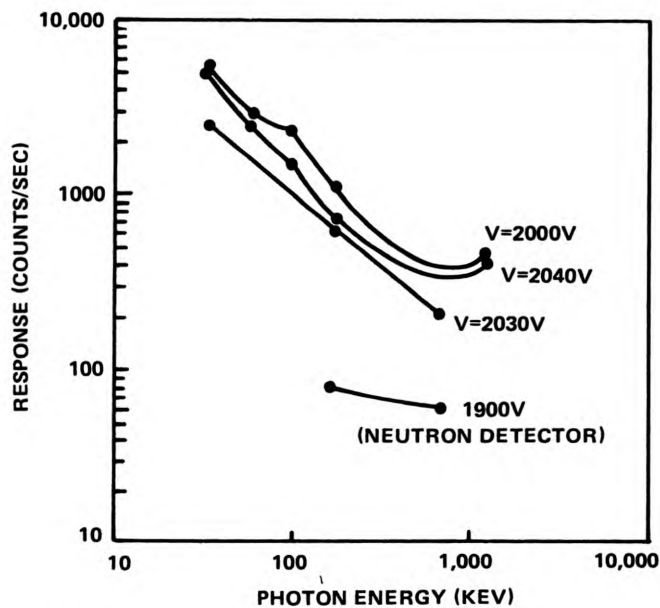


FIGURE 7.16 RESPONSE OF FOUR COMPARABLE AVALANCHE DIODE DETECTORS TO DOSE RATES OF 1 R/hr OF RADIATION OR 1 rad/hr—NEUTRON RADIATION AS A FUNCTION OF ENERGY (AFTER MOLDOFSKY, 1972).

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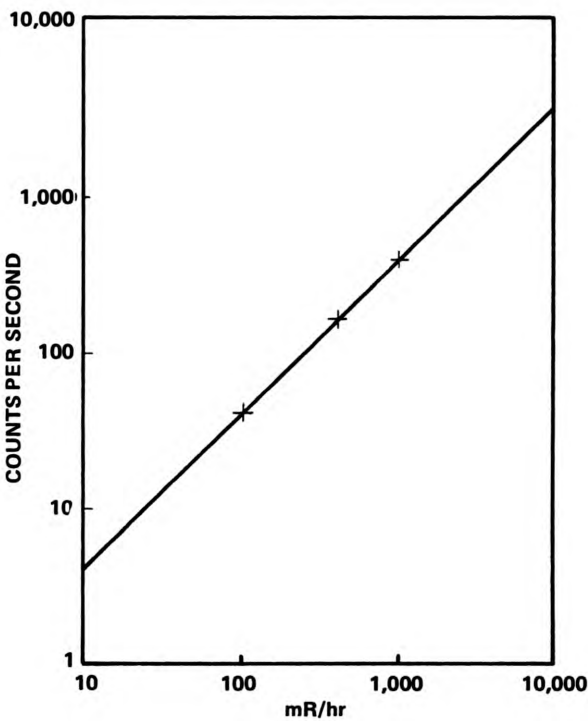


FIGURE 7.17 MEASURED LINEARITY BETWEEN PHOTON DOSE RATE AND COUNT RATE MEASURED AT 662 KeV (MOLDOFSKY, 1972).

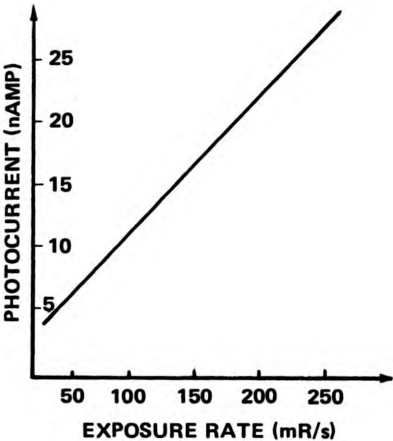


FIGURE 7.18 DIODE PHOTOCURRENT VS. EXPOSURE RATE FOR A SILICON P-I-N PHOTODIODE (LANZA, 1979).

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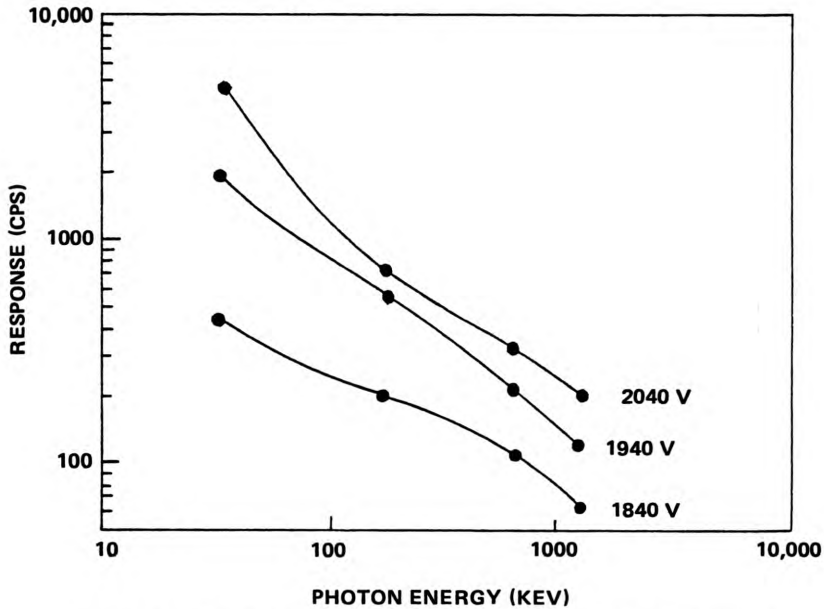


FIGURE 7.19 EFFECT OF REDUCED DETECTOR BIAS ON PHOTON RESPONSE (MOLDOFSKY, 1972).

7.3.2 Calibration for Spectroscopy Purposes

To calibrate a semiconductor system intended for spectroscopy, a knowledge of the source type (i.e., beta, alpha, gamma, etc.), fluence, and geometry must be available. For this discussion, it is assumed that a Ge(Li) detector is being set up for high resolution analysis of samples containing gamma ray emitting sources of many varying energies, necessitating the use of a multichannel analysis (MCA) system.

Modern MCA systems are non-portable assemblies (portable systems are available, however, for limited purposes) featuring CRT monitors and "hands on" computer control. Full calibration for spectroscopy consists of system linearity, resolution, energy and efficiency calibrations.

Energy Calibration and Pulse-Height Scale

With any gamma-ray energy analysis system the essential problem is the ability to establish a precise energy vs pulse-height scale. Checks of linearity between pulse-height and energy are made using monoenergetic gamma sources of precisely known energies. Figure 7.20 shows a typical calibration curve where the variation in linearity is very low over a range of 4000 channels.

After non-linearity of response checks have been completed, the actual gamma energy vs channel number calibration using the

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chosen gain setting should be performed with sources emitting gamma-rays of accurately known photon energies. Table 7.1 lists the gamma and x-ray sources available as standard reference materials supplied by the NBS. Figure 7.21 is a Ge(Li) spectrum of a typical gamma-ray source used for the energy versus channel calibration.

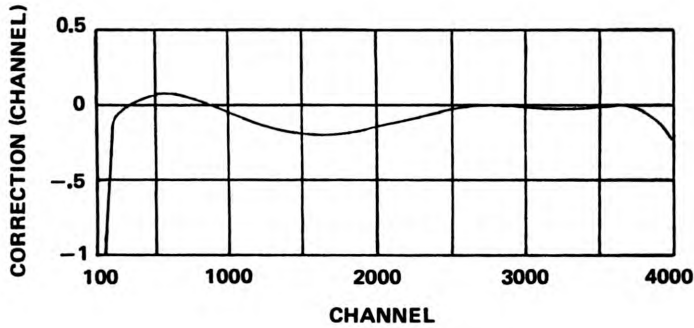


FIGURE 7.20 MEASURED INTEGRAL LINEARITY OF A 4096-CHANNEL GAMMA-RAY-SPECTROMETER SYSTEM (HEATH, 1969).

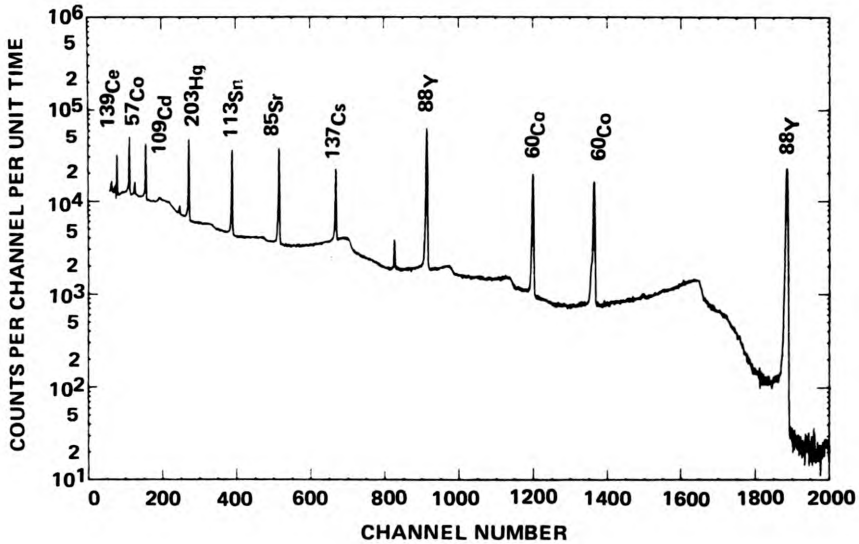


FIGURE 7.21 GAMMA-RAY SPECTRUM OF AN NBS MIXED-RADIONUCLIDE GAMMA-RAY EMISSION-RATE STANDARD. THIS 50-ml SOLUTION SOURCE WAS COUNTED AT 5 cm FROM A 60-cm³ Ge(Li) DETECTOR (COURSEY, 1976).

Figure 6.11 in Chapter 6 shows the main pulse amplifier and analog-to-digital converter (ADC). These two subunits contain the electronics that must be adjusted to establish the desired energy range and electronic calibration of the system and are adjusted in

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TABLE 7.1 PHOTON POINT—SOURCE STANDARD REFERENCE MATERIALS (SRM's)
AVAILABLE FROM THE NATIONAL BUREAU OF STANDARDS (NCRP 1958).

GAMMA—RAY SOURCES ^a			
PARENT RADIONUCLIDE	HALF LIFE	PHOTON ENERGY KeV	PHOTON PROBABILITY PER DECAY ^b
241Am	443±2y	26.35	0.0258±0.0022
		59.54	0.363 ± 0.004
109Cd ^c	453±2d	88.04	—
57Co ^c	270.9±0.6d	122.06	0.8559±0.0019
139Ce ^c	137.65±0.05d	165.85	0.8006±0.0013
203Hg ^c	46.59±0.05d	279.19	0.815 ± 0.008
51Cr	27.704±0.002d	320.08	0.0980±0.0010
113Sn ^c	114.9±0.1d	391.69	0.6490±0.0020
85Sr ^c	64.85±0.03d	513.99	0.980 ± 0.010
207Bi	38±3y	569.67	0.978 ± 0.005
		1063.62	0.74 ± 0.03
		1770.22	0.073 ± 0.004
137Cs ^c	30.0±0.2y	661.65	—
94Nb	(2.03±0.16) x 10 ⁴ y	702.63	1
		871.10	1
54Mn	312.5±0.5d	834.83	0.999760±0.000002
88Y ^c	107±1d	898.02	0.934 ± 0.007
		1836.04	0.9935±0.0003
65Zn	244.1±0.2d	1115.52	0.5075±0.0010
60Co ^c	5.271±0.001y	1173.21	0.99900±0.00020
		1332.46	1
22Na	2.602±0.002y	1274.54	0.99940 ± 0.00020
X RAY SOURCES			
55Fe	2.7±0.1y	5.89(K α_2), 5.90(K α_1), 6.49(K β)	
85Sr	64.85±0.03d	13.34(K α_2), 13.40(K α_1), 15(K β)	
109Cd	453±2d	21.99(K α_2), 22.16(K α_1), 24.9(K β)	
125I	60.14±0.11d	27.20(K α_2), 27.47(K α_1), 31(K β), 35.46(γ)	
GAMMA—RAY SOURCES OF RADIONUCLIDES WITH COMPLEX DECAY SCHEMES			
75Se	120±1d		
110mAg	250.8±0.3d		
152Eu	13.6±0.2y		
228Th ^d	1.91313 ±0.00088y		

^a In order of increasing gamma—ray energy.

^b Photon probabilities per decay are not given for those SRM's which are certified in terms of photon emission rate.

^c A mixed radionuclide source containing these nine radionuclides is distributed on annual basis (Cavallo et al, 1973).

^d This is primarily used as a standard because of the 2.6125—MeV gamma ray from the 208Tl daughter.

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conjunction with each other. The main amplifier has one or more gain or amplification level controls that will set the position of the peak in the spectrum. The ADC may contain similar controls to set the electronic resolution (conversion gain) and to "fine-tune" energy versus pulse-height calibration of the system. Basically, the actual calibration procedure is a matter of "trial-and-error." First a calibrated source having a spectrum similar to the one shown in Figure 7.21 is placed near the detector. The source should be located in a standard position with a geometry identical to that used in the actual sample analysis operation. Experimental analysis runs are then performed and adjustments are made in the gain settings until the desired correspondence between energy line and channel number is achieved. System linearity checks and/or corrections for system nonlinearities need to be made with widely varying source energies through the energy range of interest.

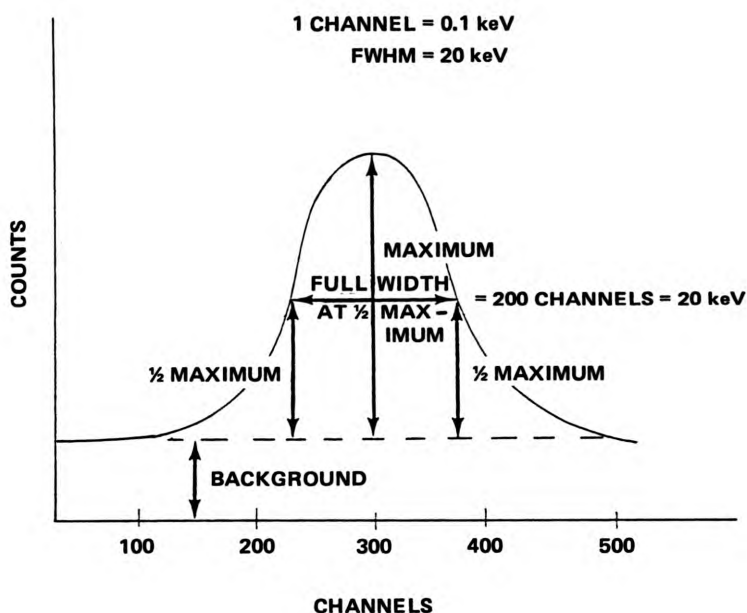


FIGURE 7.22 FULL WIDTH AT HALF MAXIMUM.

Resolution Determination

Most methods used to determine the concentration of a particular nuclide present in a sample or to help deconvolute a multi-element spectrum require that peak width versus energy determination be made. Generally, the full-width-at-half maximum (FWHM) resolution measurement is performed by: 1) subtracting the background count from each peak channel; 2) finding the maximum counts at the center of the peak; and, 3) finding the width of the

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peak, in channels, at one half the maximum count. The number of channels can be converted directly into keV (see Figure 7.22). This procedure is generally performed manually; however, there are computer programs available to produce resolution versus energy values automatically. Figure 7.23 is a sample of a typical curve obtained from a Ge(Li) detector system. The spectrum of Figure 7.21 could be used for this determination. Values taken from this curve can then be used to determine the integrated number of counts under the peak using suitable algorithms. This procedure will, however, produce erroneous values for annihilation photon peaks at 511 and 1022 keV which are broader than would be predicted by the resolution versus energy calibration curve.

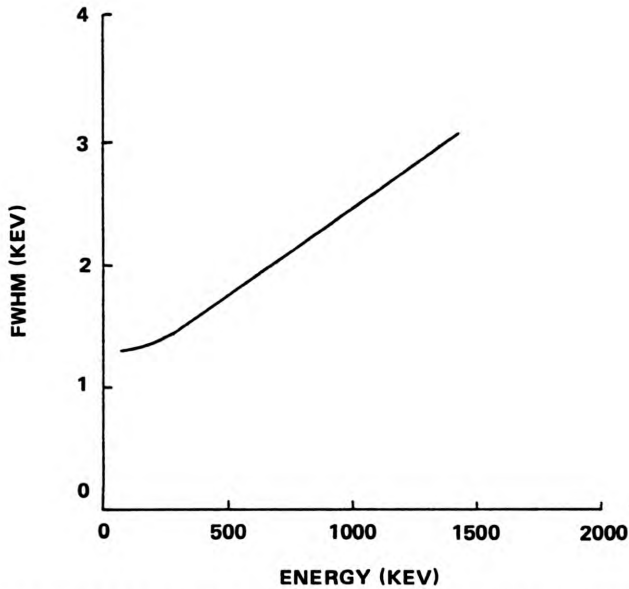


FIGURE 7.23 TYPICAL RESOLUTION CURVE FOR GAMMA PHOTONS IN A Ge(Li) DETECTOR SYSTEM.

Detector Efficiency Calibration

The efficiency of a Ge(Li) detector is usually given relative to a NaI(Tl) crystal detector. This efficiency may be misleading since the relative comparison is made using the 1.33 MeV gamma peak of ^{60}Co from a 3 x 3 inch NaI(Tl) crystal at 25 cm from the source. The absolute efficiency of the NaI(Tl) under those conditions is 1.2×10^{-3} and the relative Ge(Li) efficiency is obtained from:

$$E_r = \frac{C_t}{1.2 \times 10^{-3} C_s} \quad (7.2)$$

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where E_r is the relative efficiency of the Ge(Li) detector, C_t is the total number of counts under the 1.33 MeV peak in a period of time, and C_s is the number of gamma rays emitted by the source in the same period of time. Efficiency values of about 10 percent or greater (relative to the NaI(Tl) standard) are typical for Ge(Li) systems. This means little, however, to a user measuring low energy photons in the range of 50-1000 keV.

The absolute efficiency of the detector versus energy is almost universally used to describe the sensitivity characteristics of a system. This absolute efficiency calibration is performed using secondary standards from NBS (or other suppliers) of radiation sources. Efficiency measurements must be performed on standard geometries with the source in a form identical to that of the samples that are to be analyzed.

For the actual calibration, a single source consisting of many nuclides possessing widely varying gamma lines may be used. Otherwise, many individual sources (one at a time) containing only a few gamma lines are placed near the detector. Dead-time considerations must be observed since they affect the system resolution at high count rates. The sample should be counted for a sufficient length of time to obtain peaks of meaningful statistical definition above background. Integration techniques are then used to obtain the number of counts beneath the peak. The number of background counts under the peak should be subtracted from this value using a figure obtained from averaging the counts in channels on either side of the gamma peak centroid. The detector efficiency may then be calculated using:

$$E = \frac{C_p}{C_s} \quad (7.3)$$

where E is the efficiency of the detector at a particular energy, C_p is the total counts under a peak minus background, and C_s is the source activity corrected for decay. Using this procedure, a curve similar to Figure 7.24 may be constructed.

Counting efficiencies are greatly affected by source detector geometry considerations. For best performance, the source should be as close to the sensitive region of the detector as possible. In many situations re-entrant or Marinelli-type containers can be used to distribute the sample evenly around the detector. Caution must be exercised when assaying multiple photon emitting sources at close source-to-detector distances as coincident summing of one or more photons may take place in the detector. This phenomenon will reduce the apparent efficiency of the detector for those photons and can produce an erroneous activity determinations.

All systems should be recalibrated regularly. Energy versus channel number calibrations need to be checked daily. Resolution should also be checked visually daily for a selected nuclide energy.

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Efficiency checks may be made at more extended periods. Statistical checks on detector and electronic systems operations using Chi-square (see Chapter 1) or other analysis techniques should be performed regularly.

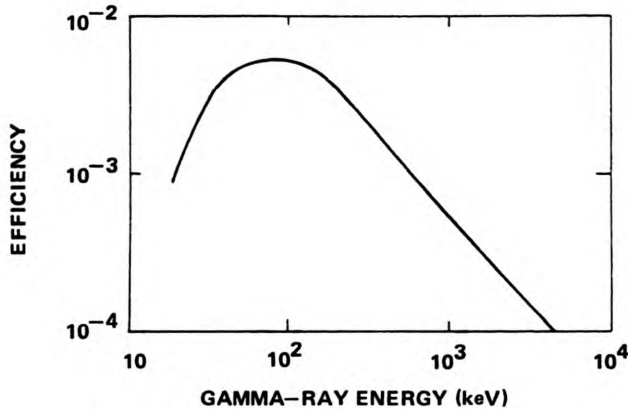


FIGURE 7.24 FULL-ENERGY-PEAK EFFICIENCY MEASURED AS A FUNCTION OF PHOTON ENERGY, FOR A 50-cm³ COAXIAL Ge(Li) DETECTOR (NCRP 58).

7.4 SUMMARY

Semiconductor materials, when used as detectors for ionizing radiation other than neutrons, can provide results unmatched by other radiation monitoring devices. Their natural advantages of generally small size and greater resolution can be, however, offset by the need for high-precision electronics, especially in spectroscopy applications. Depending on their intrinsic noise levels, some of these detectors can be used with good statistics at very low radiation flux levels. Table 7.2 summarizes the unique operating characteristics, sensitivity, resolution, and principal applications of the different classes of semiconductor detectors.

All semiconductor detectors should be calibrated using traceable sources of accurately known radiation-type and energy emission. The calibration technique is essentially the same for each detector, varying mainly due to difference in device construction, the geometries used for the calibration, and the quality of the detecting electronics.

Reliability and accuracy of measurements obtained from the detector depend to a great extent upon the quality of the calibration, the sensitivity of the detecting electronics, and the handling care before and during radiation measurement operations. The latter consideration is especially important because of the basic fragility of the detector crystal. Ruggedized detectors are available for some applications and should be chosen when possible.

TABLE 7.2 PROPERTIES OF VARIOUS CLASSES OF SEMICONDUCTOR DETECTORS.

UNIQUE OPERATING CHARACTERISTICS		SENSITIVITY	RESOLUTION	APPLICATIONS	COMMENTS
SURFACE BARRIER	Stacking for total energy absorption possible.	Good for alpha, beta, fission fragments.	Good	Particle counting Spectroscopy.	Has replaced G-M Tube in some applications; environmental shielding available; temperature sensitive.
	Pulse, current or voltage measurements.	Good for x ray gamma; good for pulse counting; possible good neutron detector	Fair	Dosimetry particle counting.	Current-measuring mode of importance in beta and photon dosimetry; temperature sensitive.
DIFFUSED JUNCTION	Low energy x ray or gamma; No LN ₂ needed.	Good for gamma and x rays less than 100 keV.	Good	Spectroscopy.	Used in x ray analysis systems; LEP detectors.
	Large sensitive region; high energy operation → 3 Mev. LN ₂ needed constantly.	Good for gamma rays greater than 100 keV.	Excellent	Spectroscopy.	
INTRINSIC MATERIALS	No LN ₂ needed until operation.	Very good for gamma greater than 10 keV.	Excellent	Spectroscopy.	Will replace Ge(Li) for most applications.
	High internal gain, low background.	Good for x rays less than 1 keV, betas greater than 5 keV; fast neutrons.	Good	Particle counting spectroscopy, Dosimetry.	Use in particle spectroscopy and dosimetry.
BULK CONDUCTIVITY DETECTORS	High stopping power; room and high temperature operation.	Good for gamma greater than 20 keV (CdTe).	Poor (CdTe Spectroscopy)	Particle counting spectroscopy, Dosimetry.	Poor material purity (CdTe) in present manufacturing processes; dosimetry of beta s and photons.

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The increase in the use of semiconductor materials to replace previously employed detection system will continue. Junction, avalanche, and bulk conductivity devices can be expected to gain increasing popularity as particle counters and in dosimetry applications especially as personnel dosimeters. Intrinsic materials, because of their handling advantages, will become the detector of choice for photon spectroscopy.

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CHAPTER 8

SOLID STATE LUMINESCENCE DOSIMETRY

J. H. SCHULMAN

8.1 INTRODUCTION

Exposure to radiation can produce both transient effects and long lasting changes in the properties of a solid. The kind of effect produced depends on the nature of the solid, the type of radiation, and the temperature during and after the exposure. The effects of irradiation are also often dependent on the dose rate and the total dose to which the solid is exposed. An invariable transient effect, of course, is the heating of the solid - an effect which is a consequence of the thermal degradation of all of the absorbed energy or a part of it after other primary processes have taken place. We shall not be concerned with this direct heating effect, although it can be used as an absolute method of dosimetry (calorimetry) (Laughlin and Genna, 1966).

Another important transient effect of radiation is the production of prompt fluorescence. This is one of the types of luminescence or so-called "cold light," i.e., light emission that is not caused by incandescence. This directly excited luminescence is the basis for the "scintillation counter" radiation detector (Birks, 1953; Ramm, 1966), wherein ionizing radiation provokes flashes of light in an efficient luminescent material called a "phosphor" or "luminophor," and these light flashes are observed by a photodetector. This prompt luminescence can be used for dosimetry by integrating the light signal over the time of exposure. However, the important luminescence-based solid-state dosimetry methods "thermoluminescence" and "radiophotoluminescence" (or "radio-photofluorescence") (Schulman et al., 1951, 1953) employ other longer lasting phenomena, in which luminescence is a secondary rather than a primary effect of the irradiation. In favorable cases, i.e. when the dose is sufficiently large, these longer lasting radiation effects can also be visually observed as light absorption or color changes in the solid. With high doses the luminescence effects can also be seen by visual means. However, the ability of luminescence methods to detect doses of interest for area and personnel monitoring (i.e., in the milliroentgen range) rests entirely on the existence of an extremely sensitive light detector, the photomultiplier tube.

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8.1.1 Optical Methods of Solid State Dosimetry

The principal optical effects used in solid state dosimetry are listed and described in Table 8.1 (Schulman, 1965; Fowler and Attix, 1966; Amelinckx et al., 1969; Piesch, 1972; Becker, 1973; Cameron et al., 1968; Oberhofer and Scharmann, 1981; McKinlay, 1981), and the operation of the two major luminescence dosimetry methods is illustrated schematically in Figure 8.1. It will be obvious that dosimetry methods based on optical changes are relative methods which require calibration with reference exposures.

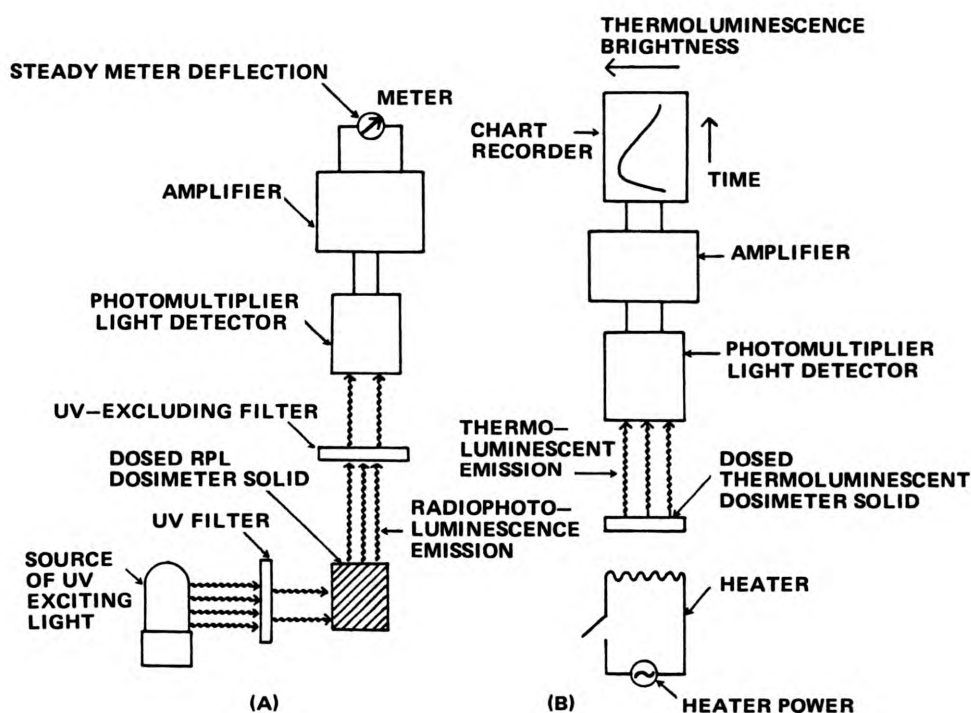


FIGURE 8.1 PRINCIPLES OF TWO MAJOR LUMINESCENCE DOSIMETRY METHODS.

(A) Measurement of radiophotoluminescence response.
(B) Measurement of thermoluminescence response.

8.2 PRINCIPLES OF RADIATION-INDUCED OPTICAL CHANGES IN SOLIDS: THE ALKALI HALIDES

The principles underlying the optical effects of Table 8.1 are closely related and are perhaps best understood by examining the pre- and post-irradiation properties of the alkali halides, which are prototypes of the insulating solids used in dosimetry (Schulman and

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TABLE 8.1 MAJOR OPTICAL EFFECTS USED IN SOLID STATE DOSIMETRY.

<u>EFFECT</u>	<u>DESCRIPTION</u>	<u>MEASUREMENT PROCEDURE</u>	<u>REMARKS</u>
Coloration	Radiation exposure produces new optical absorption in originally transparent material.	Optical absorption measurement with spectrophotometer or densitometer.	Measurement does not bleach coloration. Dose measurement is repeatable at will, and successive doses between readings are cumulative. Absorption can be "bleached" by prolonged heating.
Radiophotoluminescence (Radiophotofluorescence)	Radiation produces ability to photoluminesce under specific wavelength of exciting light in a solid originally non-luminescent under this light.	Steady-state fluorescence brightness measurement with a fluorimeter.	Measurement does not affect the luminescence ability. Dose measurement is repeatable at will, and successive doses between readings are cumulative. Effects can be erased by prolonged heating.
Thermoluminescence (Radiothermoluminescence)	Radiation produces ability to emit light when solid is warmed to temperatures lower than those needed to produce incandescence.	Measurement of luminescence as a function of time while temperature of solid is continuously raised.	Measurement procedure discharges the dose indication.

Compton, 1962). These salts (typified by ordinary table salt, NaCl, and by LiF, the most widely used thermoluminescent material) are of simple composition and structure; furthermore, their inherent and radiation-induced optical properties have been extensively studied. We will treat their absorption properties first, then the closely associated property of radiophotoluminescence, and finally the phenomenon of thermoluminescence.

8.2.1 Absorption and Radiophotoluminescence: the F-band as a Prototype

The basic optical properties that will first concern us are shown schematically in Figure 8.2. Curve (a) represents the pre-irradiation (i.e. "intrinsic") absorption spectrum of a pure alkali halide; this absorption is typically located in the "far" or short-wavelength ultraviolet region of the spectrum, and it is not affected by exposure to high energy radiations. Curve (b) is the principal new absorption produced when the crystal is irradiated with high energy radiation; it generally peaks in the visible spectrum and has weaker components towards the "near" (longer wavelength) ultraviolet. The principal absorption band is called the "F band" (from the German: Farbe = color, because it gives the normally colorless crystal a visible color), and the weaker accompanying absorption bands are given the arbitrary designations K, L₁, L₂ ...etc. Curve (c) is the spectral distribution of the light emission provoked by illuminating the irradiated crystal with light absorbed in the F band (or the K, L..., bands). This luminescent emission is called a "photoluminescence" because it is provoked ("excited") by light.

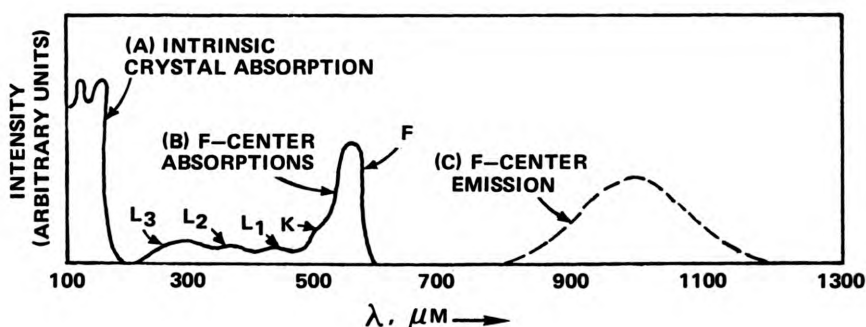


FIGURE 8.2 SCHEMATIC REPRESENTATION OF ABSORPTION AND LUMINESCENCE SPECTRA OF AN ALKALI HALIDE CONTAINING F CENTERS.

Under conditions where the radiation-induced F band is stable, the growth of the F band with dose could be used as a method of

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dosimetry; this would be the prototype of absorption-change dosimetry. Moreover, under conditions where illumination into the F band absorption produces the luminescence emission shown in (c), the luminescence intensity could also be measured and related to dose: this would be the prototype of radiophotoluminescence dosimetry. (This cumbersome name signifies the radiation-induced ability to photoluminesce; luminescence excited directly by radiation, as in the scintillation counter, is called "radioluminescence.") It is characteristic of both dosimetry methods, as noted in Table 8.1, that the act of "reading" the dose (measuring the optical absorption, or measuring the luminescence brightness under some standard intensity of exciting light) does not erase the radiation-induced absorption nor change the luminescence intensity. Dosimeters operating on either basis could, in principle, be re-read as many times as one wished without erasing the dose indication, and additional doses could be accumulated and integrated even if intermediate readings are made.

8.2.2 Crystal Structure and Energy Bands in Ideal Crystals

We will now examine the structure and properties of the alkali halides at the atomic level to see how the F band and other absorption bands arise and what atomic-scale changes are involved in their absorption and luminescence. The structure of these salts consists of two interpenetrating cubic lattices of alkali and halide ions, as shown in two-dimensional representation in Figure 8.3(a). The ideal crystal consists of an uninterrupted alternation of alkali (+) and halide (-) ions as shown in the Figure.

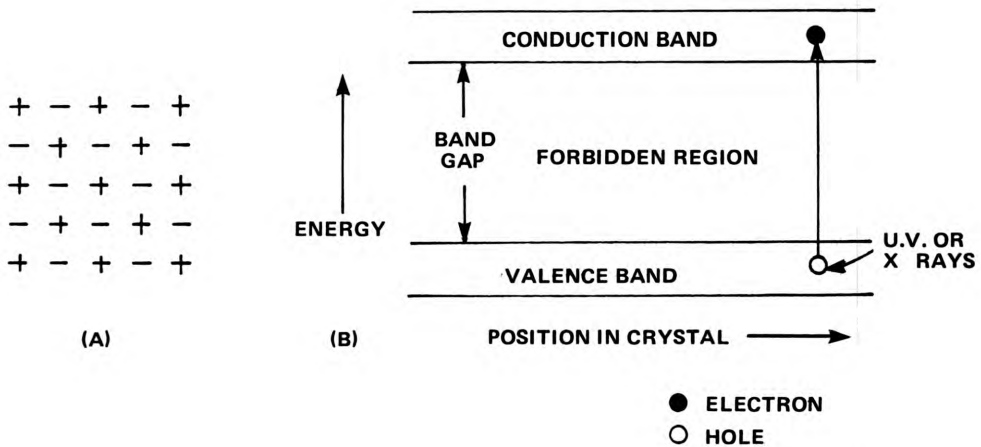
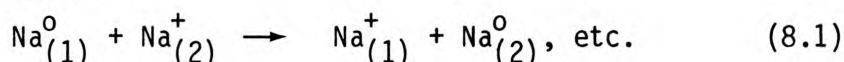


FIGURE 8.3 (A) SCHEMATIC STRUCTURE OF IDEAL ALKALI HALIDE CRYSTAL IN TWO DIMENSIONS.
(B) ENERGY BAND STRUCTURE OF AN INSULATOR CRYSTAL.

Figure 8.3(b) is a so-called energy level diagram which depicts the energy "bands" which electrons may occupy throughout such an ideal crystal. The allowed electronic energy levels in crystals are related to the allowed levels of the electrons in the individual atoms or ions that compose the crystal. In the isolated atoms the allowed energy levels are very sharply defined and are separated by broad regions of forbidden energies. When the atoms are brought together to form the crystal, however, these sharp levels are broadened into quasi-continuous "bands" of allowed energies, due to the interaction of the atoms with each other and the periodic potential field of the crystal. As in the isolated atoms, however, the allowed energy bands are separated by regions of forbidden energy, the "band gap."

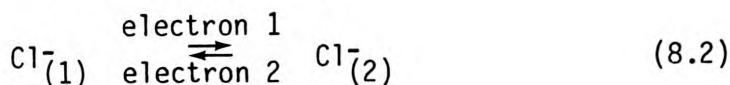
Valence Band and Conduction Band: Electron and Hole Conduction

In the alkali halides, the lower band shown in Figure 8.3(b), the so-called "valence band," arises from an energy level of the halide ions (Cl^- , F^- , etc.) that compose the crystal. All the available energy states in that band are occupied by electrons which belong to the halide ions, i.e. the band of energies is completely filled. The upper band of allowed energy in Figure 8.3(b), the so-called "conduction band," arises from energy levels of the alkali atom (i.e. Na^0 , Li^0 , etc.). Since the salt crystal consists of alkali ions, Na^+ , Li^+ , etc., not alkali atoms, the conduction band contains no electrons, i.e., it is normally empty or unoccupied. However, if an electron were somehow injected into the crystal or stripped off of a Cl^- ion, it could assume an energy level within the conduction band; crudely speaking this would correspond to one of the alkali ions of the crystal being changed to an alkali atom: $\text{Na}^+ + e \rightarrow \text{Na}^0$, (where e is the electron). Since all alkali ions of the ideal crystal are equivalent, the electron could move from one alkali to another with no change in energy, and hence it could migrate through the crystal:



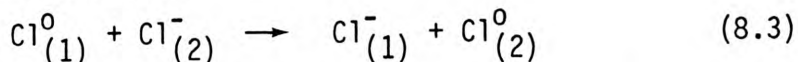
Thus, the initially non-conducting crystal acquires electrical conductivity because the electron charge carriers can physically move through the crystal when they possess sufficient energy to be in the conduction band. (This statement is generally shortened to "the electrons move through the conduction band.")

In the unperturbed crystal interchange of electrons between different halide ions, (i.e., interchange of electrons at different positions in the valence band) represented by:



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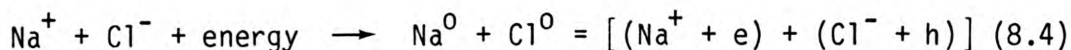
will result in neither a net change of energy nor a net movement of electrical charge. But if an electron is removed from a halide ion to transform it into a halogen atom ($\text{Cl}^- \rightarrow \text{Cl}^0 + e$), the atom will be the site of an electron deficiency; this site will therefore have a local positive charge which is called a "positive hole" or just a "hole" (h). Because of the equivalency of all the halide ions in the crystal, an electron from one of these ions could be transferred into the "hole," corresponding to:



This could be followed by further exchanges with other Cl^- ions, so that the hole in the valence band, a carrier of positive charge, could wander through the crystal. Electrical conductivity can thus be imparted to the crystal, in this case because of charge transport by holes in the valence band rather than by electrons in the conduction band.

Optical Absorption and Luminescence Processes

In many materials (semiconductors) the energy gap between the valence and conduction bands is fairly small, so that a relatively small input of energy to the crystal suffices to transfer an electron from the valence band to the conduction band; thermal energy from room temperature or from moderate heating of the semiconductor or energy provided by absorption of infrared light is sufficient. But in insulators such as the alkali halides the energy gap is quite large, and even heating to several hundred degrees will not cause electron transfer or "excitation" to the conduction band. To excite an electron to the conduction band in insulators the absorption of energetic radiation such as far ultraviolet light, x rays, or other high energy radiations is required as shown in Figure 8.3(b). The absorption of these radiations, in effect, produces the following reaction:



The electron and hole can then wander through the crystal in their respective energy bands until they meet and recombine, thus de-exciting the crystal either by emitting their energy as luminescence ("recombination radiation") or by a non-radiative process which simply dissipates the energy as heat and warms the crystal.

8.2.3 Imperfections in Real Crystals

Formation of Localized Centers and Energy Levels

In real crystals, however, many other possible fates await the electrons and holes freed by radiation. Some of these alternate

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fates are the basis of the optical effects of interest for dosimetry. Extensive studies have shown that real crystals do not attain the ideal structure shown in Figure 8.3(a), but that they contain a great many localized structural imperfections or "defects" of the types illustrated in Figure 8.4(a) through 8.4(c). The imperfections

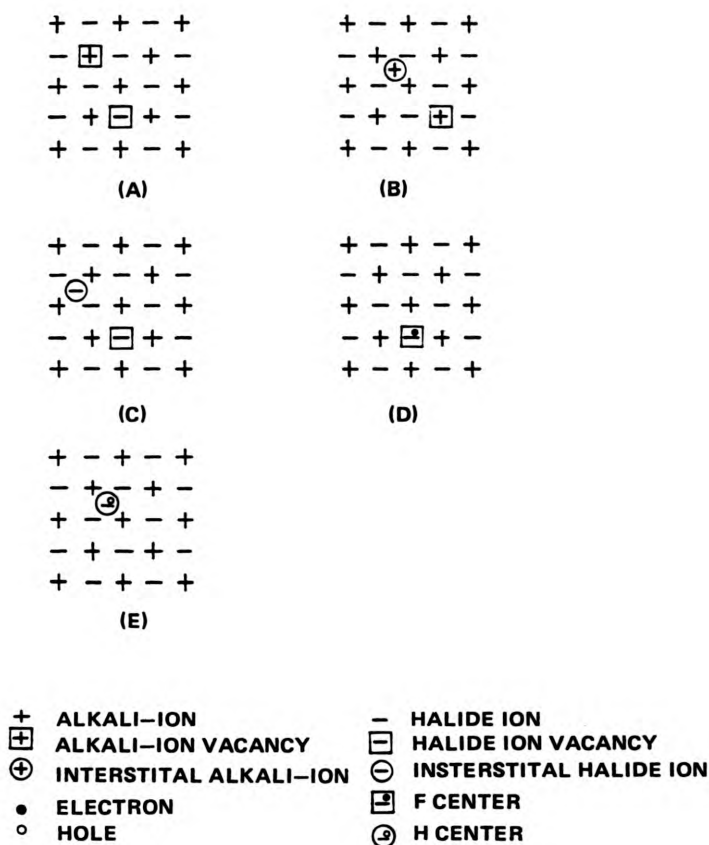


FIGURE 8.4 Schematic representation of some structural imperfections as well as trapped-electron and trapped-hole color centers, in alkali halides. The particular center consisting of an electron trapped at a halide-ion vacancy shown in part (D) of Figure 4 is called an "F" center (after the German word, "Farbe"). The specific trapped-hole center shown in part (E) of Figure 4 is the so called "H" center, which is formed when alkali halides are irradiated at a very low temperature. For most alkali halides the principal F-center absorption band lies in the visible region of the spectrum: the H-band absorption lies in the near ultraviolet.

normally found in the alkali halides are of the type shown in Figure 8.4(a), where occasional positive and negative ions are missing at random throughout the crystal. In a pure crystal the number of missing positive ions (positive-ion "vacancies" V+) must equal the number of negative ion vacancies (V-), because the crystal as a whole

must be electrically neutral. Other imperfections are possible, in principle, such as those shown in Figure 8.4(b) and 8.4(c), in which alkali or halide ions, respectively, have been displaced from normal lattice positions into interstitial positions, leaving behind corresponding vacancies. It is important to note that the concentrations of crystal defects depend very strongly on the thermal and mechanical history of the crystal and on the presence of even trace amounts of chemical impurities, which are crystal defects in and of themselves.

The existence of imperfections has important consequences when the crystal is exposed to ionizing radiations. This results from the fact that the halide-ion vacancies (Figure 8.4(a) and 8.4(c)) are regions of localized positive charge, because the negative ion that normally occupies the site is missing and the positive charges of the surrounding alkali ions are not fully neutralized. If an electron is freed in the crystal by action of ionizing radiation and wanders near the halide ion vacancy, it is attracted by a Coulomb force to the localized positive charge, and it can consequently be "trapped" in the vacancy as shown in Figure 8.4(d). In a similar way the interstitial halide-ion shown in Figure 8.4(c) is a region of localized excess negative charge. When radiation strips an electron from one of the the normal lattice ions, the hole can migrate through the crystal and be attracted and bound Coulombically as shown in Figure 8.4(e). It should be noted that ionizing radiations can also produce new vacancies and interstitials in structurally perfect regions of the crystal, in addition to liberating free electronic charges that can act with these defects in the manner just described.

Optical Properties of Trapped Electron and Trapped Hole Centers

It has been shown that the atomic system or "center" comprising an electron trapped at a halide-ion vacancy (Figure 8.4(d)) is responsible for the F band (and its associated K and L bands) in the alkali halides; this defect center is therefore called the "F-center". The F-center roughly resembles a hydrogen atom, in which the electron is bound by the positive charge of a proton. Like a hydrogen atom the electron-vacancy system has discrete allowed energy levels and can make transitions between these levels by absorption or emission of light quanta (photons) of the proper energy. The F-center system differs from a free hydrogen atom, which has very sharply defined energy levels, in that the F-center electron feels the effects of the crystal's ions and their thermal vibrations, and the allowed levels are consequently considerably broadened. Instead of the spectral "line" absorption of the hydrogen atom, the F-center absorbs in a relatively broad spectral band of light. The absorption of a light photon within this spectral band can raise the F-center electron from the very lowest ("ground") energy state of the system to one of its higher ("excited") energy states, and a sufficiently energetic photon (corresponding to absorption in an L

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band) can even expel the electron from its vacancy "trap" altogether, a process which resembles the photoionization of a hydrogen atom. The return transition of the F-center electron from its lowest excited state to the ground state is responsible for the broad-band F-center luminescence.

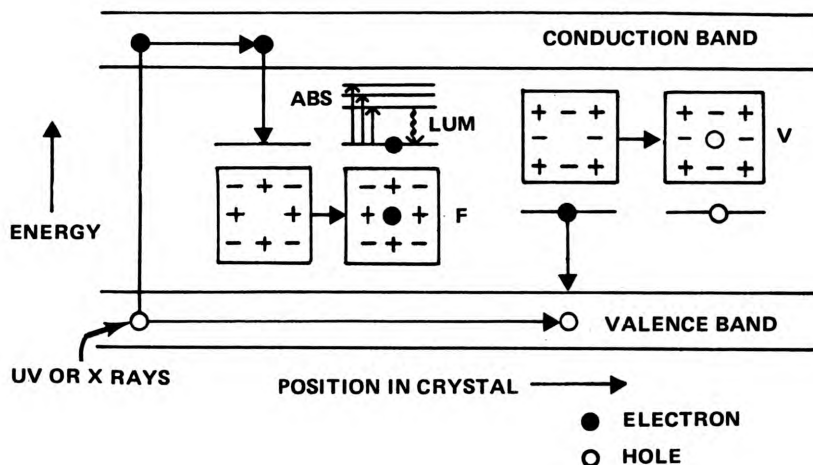


FIGURE 8.5 REPRESENTATION OF F- AND V-CENTER FORMATION DURING IRRADIATION OF AN ALKALI HALIDE, IN TERMS OF A SIMPLIFIED ENERGY-LEVEL DIAGRAM.

Figure 8.5 illustrates how the localized defects we have been discussing introduce localized energy levels into the forbidden band, and how these levels are involved in the formation of radiation-induced F-centers; it also illustrates the absorption and luminescence properties of these centers. The discrete unoccupied level created by a halide-ion vacancy is shown as a short horizontal line above the picture of the vacancy. On capture of an electron from the conduction band, the vacancy is converted into an F-center, also shown schematically in the figure. The ground and excited states of the F-center are indicated as localized energy levels above the schematic picture of the center; and the optical transitions which give rise to the F, K, L... absorption bands are indicated by vertical arrows pointing upward. The radiophotoluminescence, the luminescence emitted after optical excitation of the F-center, is shown by the wavy arrow pointing downward and connecting the lowest excited state with the ground state. The figure also indicates one of the possible fates of the hole produced in the valence band. The example shown is the formation of a hole trapped at an alkali-ion vacancy to produce a so-called "V-center," the "antimorph" of the F-center. Trapped-hole centers, like trapped-electron centers, have their own unique optical characteristics and transitions, but our emphasis will continue to be on the F-center in order to simplify the exposition.

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8.2.4 Special Impurity Centers: Ag^+ Ions

In the above description of the radiophotoluminescence of the F-center, we have concentrated on its behavior as a luminescence "activator." In this role the F-center is not destroyed by the act of observing or measuring it, but continues to "load and fire" (i.e., absorb light and emit luminescence) as long as the exciting light is shone upon it. Since this ideal behavior of alkali-halide F-centers as luminescence activators can take place only at low temperatures, their radiophotoluminescence is not useful for practical applications, but other centers have been discovered which can perform in this fashion at ordinary temperatures. The most important of these are centers formed by the action of radiation on solids containing silver ions, Ag^+ , incorporated in solid solution.

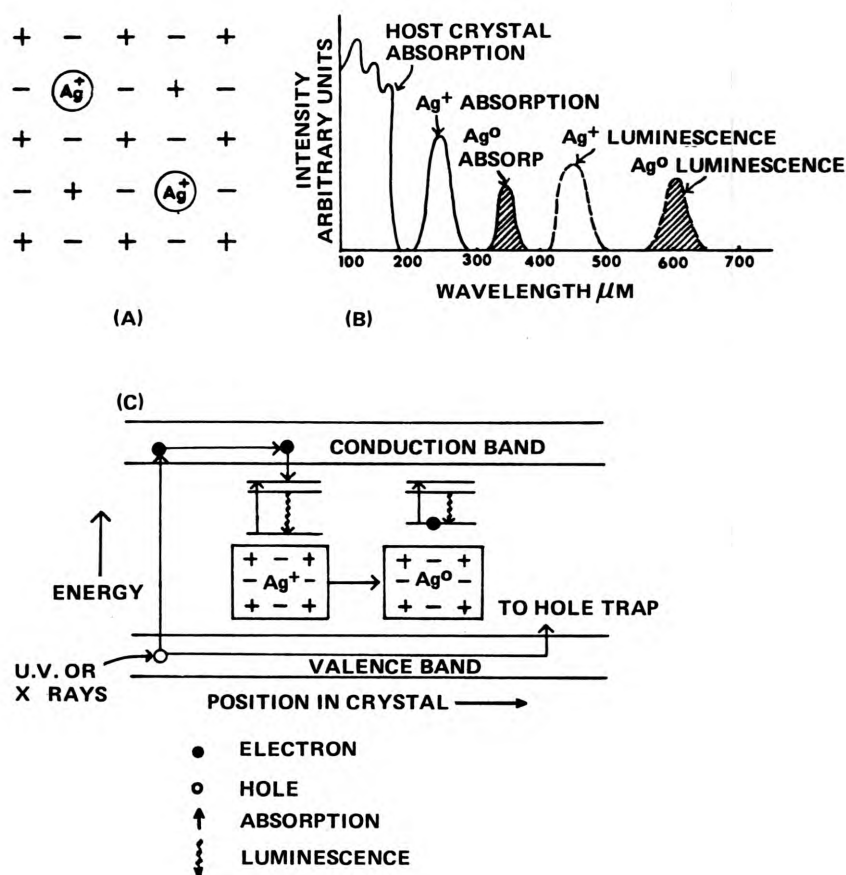


FIGURE 8.6 SCHEMATIC CRYSTAL CONFIGURATION AND RADIATION-INDUCED CHANGES IN ALKALI HALIDE CONTAINING Ag^+ -ION IMPURITY. (A) Two dimensional representation of crystal containing Ag^+ ions. (B) Absorption and luminescence spectra. (C) Representation of formation of Ag^0 centers by irradiation.

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Radiophotoluminescence of Ag^+ -Activated Alkali Halides

The picture of an alkali halide containing a small amount of Ag^+ ion impurity is shown in Figure 8.6(a), and its pre- and post-irradiation optical properties are schematically indicated in Figure 8.6(b). The important radiophotoluminescence properties of the Ag^+ ions are three-fold: (1) their chemical characteristics make them deeper, more stable traps for electrons and holes than the native crystal defects (i.e., the positive and negative ion vacancies); (2) the centers formed by electron and hole trapping, Ag^0 and Ag^{2+} , have new absorption bands in the near ultraviolet, separate from both the intrinsic crystal absorption and the absorption bands of the original Ag^+ ions; and (3) illumination into these new absorption bands at normal temperatures produces a photoluminescence (orange color) which is also different from the (blueish) photoluminescence of the original Ag^+ ions (Etzel and Schulman, 1954; Delbecq et al., 1963). By incorporating Ag^+ ions into an alkali halide in sufficient

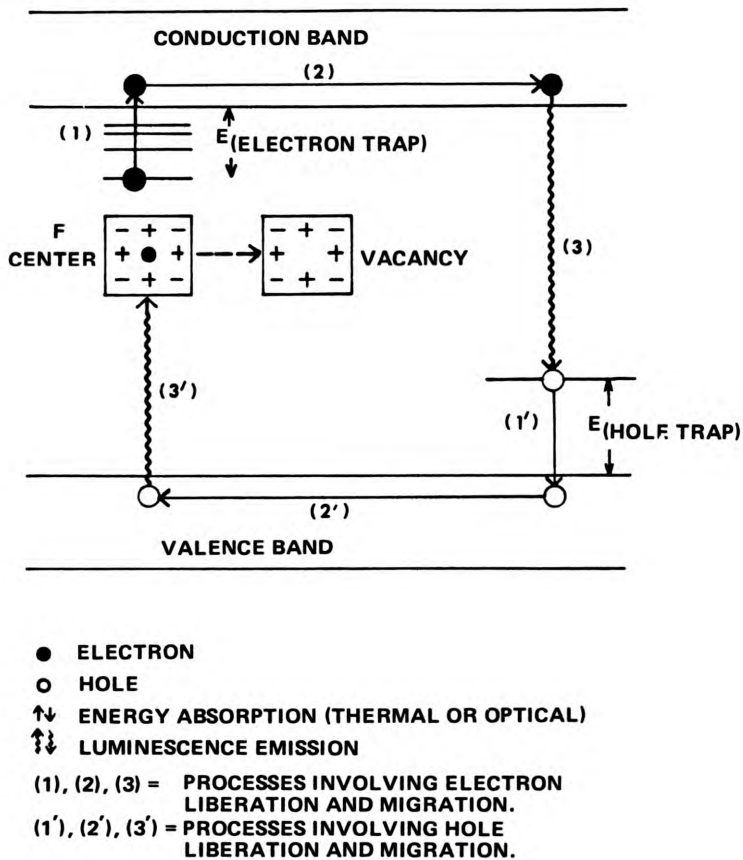


FIGURE 8.7 THERMOLUMINESCENCE AND OPTICALLY STIMULATED LUMINESCENCE.

amount, therefore, F-center formation is suppressed and the stabler luminescent centers of the Ag^0 and Ag^{+2} types are preferentially formed upon irradiation. The energy level diagram and the processes which take place are shown schematically in Figure 8.6(c), where only the formation of Ag^0 is given as an example. In practical dosimetry systems, for economic and other reasons, the host materials for the Ag^+ are special aluminophosphate glasses instead of alkali halides, but the radiation-induced changes in these glasses are similar to those illustrated in Figure 8.6 and lie in the same spectral regions (Schulman et al., 1951 and 1953; Yokota and Imagawa, 1967).

8.2.5 Thermoluminescence

Charge Trapping and "Un-trapping"

Our emphasis up to this point has been on the special optical properties of the new radiation-induced species, F-centers or Ag^0 centers in particular, formed by trapping of free electrons by halide-ion vacancies or an impurity ion, Ag^+ . Thermoluminescence dosimetry, however, is not based on the optical properties of these centers but rather on the energetics and kinetics of their formation, i.e., the trapping and "un-trapping" processes. If an electron is freed from its trap by absorption of the energy E_{trap} as shown in Step 1 of Figure 8.7, it can wander through the crystal (step 2) until it encounters a trapped hole, with which it can recombine (step 3). When the recombination produces light, the luminescence is called thermoluminescence if the electron has been freed from the F-center by warming the crystal*; it is called "stimulated luminescence" if the electron ejection has been accomplished by illumination with light absorbed in the F-center absorption bands. (This stimulated luminescence is not to be confused with the stimulated emission of coherent light that takes place in lasers, a subject that lies outside of our discussion.)

A symmetrical series of processes can also occur in which a hole is optically or thermally liberated from its trap and migrates through the valence band until it encounters an electron with which it can recombine; these analogous processes are shown in steps 1', 2', and 3' in Figure 8.7. In both thermoluminescence and stimulated luminescence the trapped center (F center or trapped hole center) is

* "Delayed" light emission, or so called "phosphorescence," can often be observed after removing a solid material from the exciting source of energy. In terms of Figure 8.7, this phosphorescence arises because the ambient temperature is sufficient to provide the thermal energy needed to expel trapped electrons from their traps. Phosphorescence can thus be viewed as a thermoluminescence that takes place at constant temperature.

destroyed. In dosimetry by thermoluminescence and stimulated luminescence the reading can therefore be made, in principle, only once; the act of reading discharges the accumulated radiation-induced centers. (Special techniques for repeat readings of thermoluminescent dosimeters are possible under certain special circumstances. These special cases do not contradict the general principle stated above.) We will concentrate on thermoluminescent systems in the following discussions since there are no practical stimulated dosimetry systems in existence.

Impurity Effects on Traps and Thermoluminescence Efficiency

In order for a material to be useful for thermoluminescence dosimetry it must contain an ample and reproducible concentration of electron (or hole) traps, and when the trapped charge carriers are expelled thermally from the traps they must recombine with the charge carriers of the opposite sign by a radiative rather than a dissipative process. Furthermore, the trapping centers must be quite deep (i.e., E_{trap} must be quite large), for if the traps are shallow, thermal energy even at room temperature will be sufficient to empty them, leading to partial or total loss of the dose indication on storage (an effect called "fading"). The pure alkali halides and other pure insulator salts do not meet all these conditions, and hence the following principles have been used to develop other systems which satisfy these requirements better.

Let us first consider the requirement that the electron-hole recombination be radiative. To insure that this will happen, certain impurities that are known to be luminescent activators can be added to the salt. There is a large accumulation of experience and theory concerning these impurities from studies of phosphors used in fluorescent lighting, cathode-ray tubes, and other fields of luminescent solids. It is known that the rare-earth ions and manganese are almost "universal activators," i.e., small amounts of these ions dissolved in otherwise non-luminescent "host" materials make these materials luminesce very efficiently under ultraviolet, cathode-ray or x-ray excitation. Figure 8.8 shows what can happen when one of these activators, Mn^{+2} , is introduced as an impurity in an alkali halide. Like any other disturbance of the crystal's periodic structure, it introduces localized levels in the forbidden region. The hole which would otherwise be trapped at some imperfection native to the pure crystal, can now get trapped by (or near) the Mn^{+2} , which can be symbolized by the reaction:



When the electron is freed from the F-center by heating it will recombine with the trapped hole to reverse this reaction:



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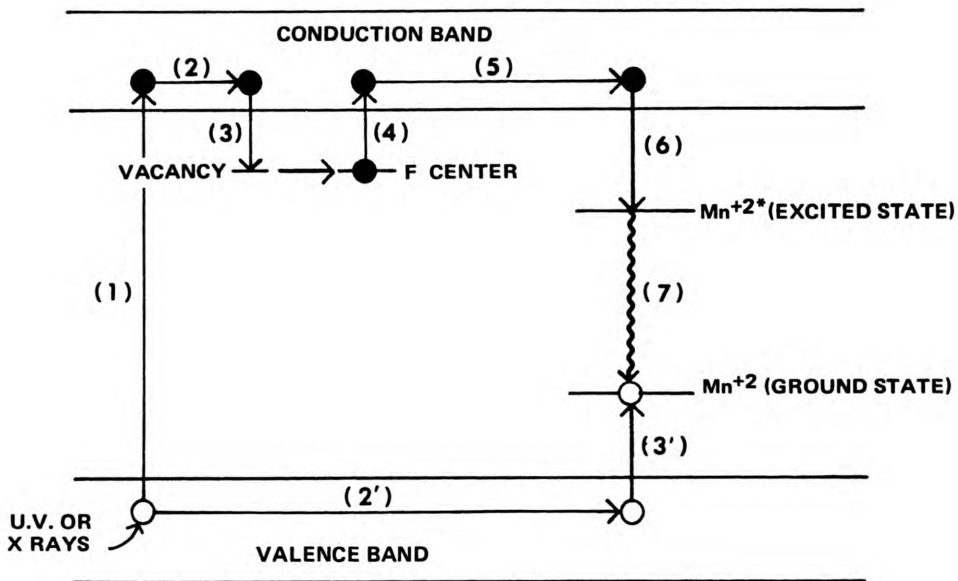


FIGURE 8.8 ELECTRON AND HOLE TRAPPING AND THERMOLUMINESCENCE IN A Mn^{+2} -ACTIVATED ALKALI HALIDE.

- (1) radiation-induced formation of free electron and hole
- (2) (3) electron trapping by negative-ion vacancy to form F center
- (2'), (3') hole trapping by Mn^{+2}
- (4) thermal ejection of electron from F center
- (5), (6) migration of electron to hole trapped at Mn^{+2}
- (7) luminescence emission from Mn^{+2}

However, the Mn^{+2} produced in this recombination is formed in an excited state (which is what the symbol * is meant to indicate), and it returns to its ground state with very efficient emission of energy in the spectral band characteristic of the Mn^{+2} activator. Thus, non-radiative or inefficiently radiative recombination centers in the pure crystal are replaced by the efficiently luminescent recombination center Mn^{+2} . The emission spectrum of Mn^{+2} varies from green to orange depending on the host crystal in which the Mn^{+2} is incorporated. Various rare-earth ion activators, emitting luminescence from the blue through the red, are also used in a similar manner.

Next, let us consider the other requirement, i.e., to build in larger and more reproducible concentrations of deep traps than one can achieve in a pure crystal. Here, again, the addition of selected impurities is made use of. The principle is illustrated in Figure 8.9, which shows how a divalent positive-ion impurity, substituting for an alkali ion in an alkali halide, affects the concentration of vacancies. For compensation of the excess positive charge of the impurity, an alkali ion must be omitted from the structure, i.e., for every substitutional divalent impurity ion, a positive-ion vacancy is created. The concentration of negative-ion vacancies is

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correspondingly decreased. Moreover, since the divalent cation impurity constitutes a local positive charge and the cation vacancy constitutes a local negative charge, the two can attract each other and form "complexes" as shown in the figure.

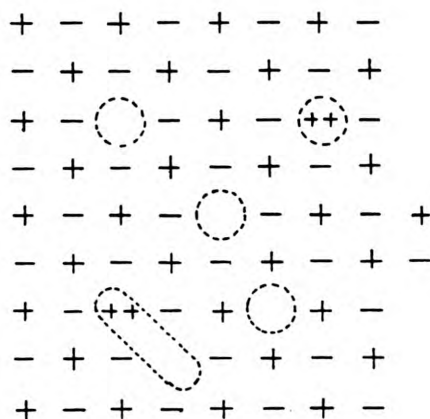


FIGURE 8.9 SCHEMATIC REPRESENTATION OF VACANCIES, SUBSTITUTIONAL DIVALENT CATION IMPURITY, AND A VACANCY-IMPURITY "COMPLEX" IN AN ALKALI HALIDE CRYSTAL.

The concentration and nature of electron and hole traps are thus profoundly affected by the impurity. Besides the changes it makes in the numbers and relative proportions of positive- and negative-ion vacancies, the divalent impurity cation is itself a possible electron trap by virtue of its excess local positive charge, and the complexes it forms also are sites where radiation-induced changes can take place more easily than at the normal lattice sites. The foregoing effects are undoubtedly involved in "dosimetry-grade" LiF, whose trapping properties are known to be connected with the presence of divalent and higher valent impurities.

The major practical thermoluminescent dosimetry materials are simple salts: lithium fluoride containing magnesium and titanium (symbolized by LiF:Mg:Ti); calcium fluoride containing manganese or rare-earths ($\text{CaF}_2\text{:Mn}$ or $\text{CaF}_2\text{:RE}$ where RE = dysprosium, samarium, thulium, etc.), and lithium borate containing manganese ($\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$). Despite their apparently simple formulae, these practical materials are actually very complex solid state systems, as can be appreciated from the above discussion and as we shall see more fully in a later section. Extensive research has not yet revealed the detailed nature of the traps and luminescent centers in these solids with certainty, although it is clear that these entities are qualitatively similar to the prototype imperfections we have described using the alkali halides as examples.

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Glow Curves

Experimental: Peak Heights and "Light Sum"

The practice of thermoluminescent dosimetry requires familiarity with another concept, the "glow curve," which is a plot of the temperature-dependence of luminescence intensity under dynamic conditions, i.e., as the temperature of a dosed material is gradually but continuously increased. As we have pointed out, the traps for electrons and holes must be relatively deep in a good thermoluminescent material; these charge carriers, therefore, will remain in their traps for long times at ordinary temperatures, and the steady-state luminescence emission at these temperatures is zero. But if the temperature is progressively raised, the trapped charge carriers will be expelled from their traps at an increasing rate and they will recombine emitting light; the rate of trap-emptying, and therefore the thermoluminescence brightness, will increase with temperature. A peak brightness will be reached, however, when the supply of trapped charge carriers decreases beyond a certain point; for although the rate of expulsion from the traps continues to increase with increasing temperature, the depletion of the store of trapped charges more than compensates for this faster expulsion rate. Finally, when all the traps have been emptied, the thermoluminescence will decrease to zero again. Further raising of the temperature will provoke only the normal incandescence that all heated bodies emit at high temperatures. The shape of the glow curve and the temperature of the "glow peak" will depend on the rate of heating and the form of the heating schedule (e.g., constant or variable rate of heating). When the brightness curve is plotted as a function of time (Figure 8.10), the area under the curve is called the "light sum," and it is determined by the total amount of energy stored in the solid in the form of trapped charges. The "light sum" is therefore proportional to the dose received by the solid and is independent of the heating details. The height of the glow peak can also be used as a measure of dose, but in this case the heating schedule must strictly reproduce the schedule used in calibrating the glow-curve reading instrument.

Complex Glow Curves: Annealing

As one can readily appreciate from Figure 8.9, most solid state dosimetry materials have more than one type of trap. The glow curves of even the best thermoluminescent solids are therefore almost never simple, but consist of many peaks. Moreover, the form of the glow curves can depend not only on the heating rate, but also on the pre-irradiation and post-irradiation treatment of the solid. Pre-irradiation heating ("annealing"), generally conducted at fairly elevated temperatures of a few hundred degrees C, will alter the distribution of trapping species in the solid by permitting diffusion, aggregation, or dissociation of the various structural

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and impurity imperfections and "complexes"; this will change the numbers and distribution of traps of different depths. A milder post-irradiation heat treatment is often used to selectively remove low-temperature glow peaks, e.g., by holding the solid at some moderate temperature until the shallow traps responsible for these peaks are emptied. In the practice of thermoluminescence dosimetry, then, the dosimeter solid must be treated thermally in a manner identical with that used in calibrating the system.

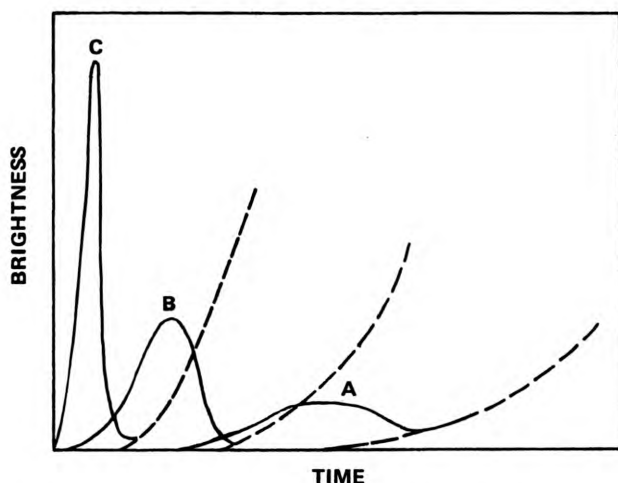


FIGURE 8.10 Thermoluminescence brightness vs time at different rates of heating, β , ($\beta_C > \beta_B > \beta_A$). — thermoluminescence; ---- thermal radiation from heater and/or phosphor. With a very rapid heating rate, the thermoluminescence can be made appear as a brief flash of light.

Theory of Glow Curves

The theory of glow curves has been studied very intensively by phosphor researchers with the aim of deducing the depths and other properties of electron or hole traps from these curves. Since we are interested primarily in the use of these curves for thermoluminescence dosimetry, an elementary discussion of their characteristics will be sufficient for our purpose (Garlick, 1949).

We consider the case where: there is only one kind of electron trap having a depth E below the conduction band (as in Figures 8.7 and 8.8); an electron freed from a trap is not retrapped but undergoes a luminescent recombination with a positive hole; the rate-determining process for luminescence is the rate of escape of the electron from the trap; and the phosphor is heated at a constant rate β°/sec . If τ is the mean lifetime of an electron in the trap, the probability

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per unit time of ejection of the electron from the trap is:

$$p = \frac{1}{\tau} = s e^{-\frac{E}{kT}} \quad (8.7)$$

where T is the absolute temperature, k is Boltzmann's constant, and s is a factor having the dimensions of frequency, related to (but generally much less) than the frequency of atomic vibrations in the solid. The luminescence intensity, I , is determined by the rate of emptying of electrons from traps, dn/dt , where n is the number of electrons in traps at a given time t :

$$I = -\frac{dn}{dt} = np = n s e^{-\frac{E}{kT}} \quad (8.8)$$

If the phosphor is heated at a constant rate

$$-\frac{dn}{dt} = -\beta \frac{dn}{dT} = n s e^{-\frac{E}{kT}} \quad (8.9)$$

whence

$$n_T = n_0 \exp \int_{T_0}^T -\frac{s}{\beta} e^{-\frac{E}{kT}} dT \quad (8.10)$$

and

$$I_T = n_0 s e^{-\frac{E}{kT}} \exp \int_{T_0}^T -\frac{s}{\beta} e^{-\frac{E}{kT}} dT \quad (8.11)$$

The graphical form of the last equation is shown in Figure 8.11 for various values of the parameters E and s at a constant heating rate of 2.5°C/sec . T^* , the glow-peak temperature, moves to higher values as E or β increase or as s decreases. The area under each curve is proportional to n_0 , the number of electrons that were initially trapped (which, in the dosimetric use of phosphors, is determined by the radiation exposure or dose). If a phosphor contains several different types of traps, with each trap characterized by its own values of E and s , the glow curve will contain several glow peaks. Those having low values of E and high values of s will not store energy well; traps having $E < 0.8$ eV along with $s \geq 10^9$ per second suffer severe depopulation in one day or less at room temperature.

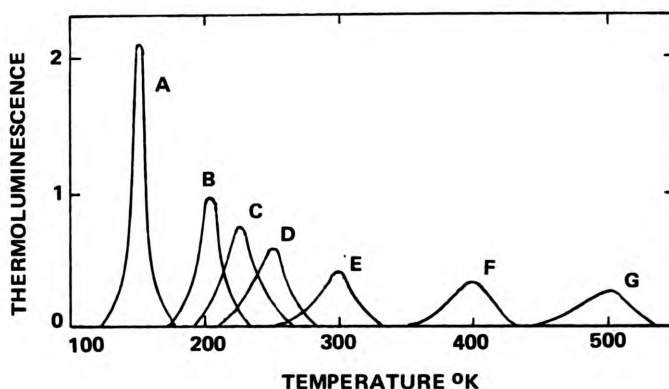
A major problem for phosphor researchers has been to determine the trap parameters from the experimental glow curves. This is rather complicated and laborious, and therefore several methods have been suggested for deducing E from the value of T^* . At T^* , $dI/dT = 0$,

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giving

$$\beta E/T^2 = s e^{-E/KT^*}$$

where E is an implicit function of T^* defined by the parameter $\Theta = \beta/S$. For $\Theta = 10^{-9}$ ("normal" values of $\beta = 10^0$ K/sec. and $s = 10^9$ per second), an empirical relation (Urbach, 1930) holds approximately as follows: E (in electron volts) = $T^*/500$. For higher values of Θ (faster heating rates or lower s factors), the denominator increases considerably; for example, a trap of 1 eV depth at a 100-fold faster heating rate ($\Theta = 10^{-7}$) would give a glow peak near 600°K rather than at 500°K .



CURVE	E(eV)	S(sec ⁻¹)
A	0.3	10 ⁹
B	0.4	10 ⁹
C	0.4	10 ⁸
D	0.4	10 ⁷
E	0.6	10 ⁹
F	0.8	10 ⁹
G	1.0	10 ⁹

FIGURE 8.11 THEORETICAL GLOW CURVES FOR PHOSPHORS
WITH A SINGLE TRAP DEPTH AND NO RETRAPPING
(GARLICK, 1949).

8.2.6 General Considerations in Applied Luminescence Dosimetry

The features desired in luminescence dosimetry systems are, of course, similar to those for other types of dosimetry: adequate sensitivity for the application in hand; uniformity of response between different dosimeters of the same type; reproducibility of response on successive readings of the same dosimeter, insensitivity of radiation-response to pre-irradiation and post-irradiation environment and history (storage times, temperatures, humidity or other atmospheres, shock, light, contamination); dose-rate independence; ease and simplicity of reading; adaptability to

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different dosimetry problems; convenient size; and low cost. Depending on the application one may add the requirement either of energy-independence or of different response to radiations of different energies or linear energy transfer (l.e.t.). The two major solid-state luminescence dosimetry systems, thermoluminescence and radiophotoluminescence dosimetry, satisfy a remarkable number of these requirements. Radiophotoluminescence was the first of these methods to achieve extensive use, initially in the form of a "casualty" dosimeter for military use, and then in a number of applications for personnel monitoring and medical dosimetry. Although it is still widely used, its popularity has been surpassed by thermoluminescence dosimetry, which is currently the most widely practiced luminescence dosimetry system. The literature on both these methods is extremely voluminous, and there are a great many commercial and specialized institutional systems based on both methods. We will discuss the most representative of these practical systems in the following sections.

Properties of Phosphors and Photodetectors

In general, the practical radiation sensitivity of a phosphor depends primarily on two fundamental properties of the solid itself and secondarily on another circumstance that is a property of the reading device rather than the phosphor. The two basic phosphor properties are: 1) the efficiency with which the absorbed high-energy radiation creates the centers of interest (either radiophotoluminescent centers or trapped electron/hole centers), and 2) the efficiency of the luminescence during the "read-out" process (i.e., on photoexcitation of the radiophotoluminescent phosphors or warming of the thermoluminescent phosphors). The secondary circumstance depends on the spectral sensitivity of the photodetector used in the reading instrument. If the radiophotoluminescent or thermoluminescent emission falls in the high-sensitivity region of the photodetector (usually a photomultiplier tube), the detector's output will obviously be higher than if the emission is in a low-sensitivity part of the detector's sensitivity. Thus, although the thermoluminescent phosphor $\text{Li}_2\text{B}_4\text{O}_7:\text{Mn}$ is intrinsically almost twice as efficient as $\text{LiF}:\text{Mg}:\text{Ti}$ in utilizing absorbed radiation to produce trapped charges, its practical efficiency is only one eighth of the latter because the blue emission of the lithium salt matches the photomultiplier tube's sensitivity better than the red emission of the borate salt does.

Environmental Effects

Luminescence dosimetry solids are not immune to disturbing environmental effects, and dosimeters are therefore designed to avoid these effects. Humidity or other chemical atmospheres can act on the surface of phosphors, dissolving and perhaps recrystallizing

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them superficially into other forms with different luminescence properties. Foreign contaminant matter can also have photoluminescent or thermoluminescent properties that could interfere with the proper measurement of the radiation-induced luminescent response. Another environmental disturbance is ambient light. It will be recalled from section 8.2 that exposure to light can have a variety of effects depending on the wavelength of the light: it can either create luminescence centers or trapped-charge centers, or "bleach" ("erase") these centers. In the first case, the light-induced centers would give a spuriously high dose indication; in the second case there would be a loss of dose-indication (light-stimulated fading). Another environmental disturbance, one that afflicts thermoluminescent phosphors rather than radiophotoluminescent phosphors, is "tribothermoluminescence," the production of trapped charges by rubbing, friction, or grinding of the phosphor. Protection of both radiophotoluminescent and thermoluminescent phosphors from all of the above effects is necessary; it is generally accomplished by proper design of the dosimeter case and by simple precautions in use.

Finally, temperature is an omnipresent but especially important environmental factor because of its intimate connection with the mobility of electrons and holes in solids, with the expulsion of charge carriers from their traps, and with the diffusion of atoms and ions in solids - all of which affect the production and stability of radiophotoluminescence and thermoluminescence centers. The effects of pre- and post-irradiation thermal treatment and storage temperatures are so varied and important that they will occur repeatedly as we discuss specific dosimetry materials in the following sections.

8.3 RADIOPHOTOLUMINESCENT DOSIMETRY SYSTEMS

Although currently superseded in popularity by thermoluminescence dosimetry, radiophotoluminescence dosimetry nevertheless retains the outstanding features that made it the first widely used solid state luminescence dosimetry system. Among these features are excellent shelf-life, ruggedness, great dynamic range (from milliR to KiloR), stability of dose indication over long periods of storage time, non-erasure of the dose indication even by repeated re-readings of the dose, and simplicity of the reading procedure. Although a lower limit of 5-10R was acceptable for the first large application of this system as a military casualty-range dosimeter, subsequent improvements in the radiophotoluminescent material and in the fluorimeter ("reader") extended the minimum detectable dose into the milliR region and made the system useful for personnel monitoring and other applications. These and other features are bringing about a revival of interest in radiophotoluminescence and making it perhaps more than competitive

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with thermoluminescence.

The basic radiophotoluminescent material is an aluminophosphate glass containing several percent of silver phosphate as a constituent. The radiophotoluminescence mechanism is similar to that described in Section 8.2 for the silver-activated alkali halides, namely, the radiation-induced change of Ag^+ ions to Ag^0 and Ag^{+2} , both species creating a new optical absorption in the near ultraviolet spectral region (Figure 8.12).^{*} Illumination of the glass with this near ultraviolet light provokes both types of centers to emit an orange luminescence (see Figure 8.6 and 8.12) whose intensity is a linear function of the dose up to a few tens of KiloR. The response of the glass is independent of dose rate from 10^{-4} R/min. to more than 10^{10} R/sec.

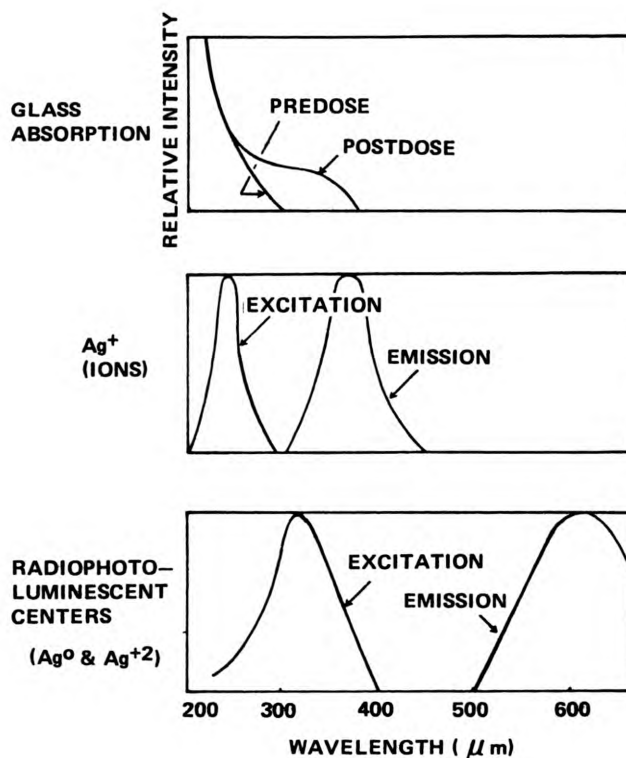


FIGURE 8.12 PRE- AND POST-DOSE ABSORPTION AND LUMINESCENCE OF SILVER-ACTIVATED ALUMINOPHOSPHATE GLASS.

^{*} There are indications that pairs or larger aggregates of Ag^0 (such as Ag_2^0 , Ag_3^0 , etc.) are also among the radiophotoluminescent centers formed in the glass; for simplicity we will continue to refer to these as Ag^0 (Yokota and Imagawa, 1967).

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Various shapes and sizes of radiophotoluminescent glass dosimeters have been used depending on the application. The most common forms are rectangular blocks of the order of a few millimeters in thickness and 1-2 cm in other dimensions. Another form, previously widely used for clinical studies, is a 1 mm x 6 mm glass rod ("fluorod"); shorter and thicker cylinders have also been used. Obviously the number of radiophotoluminescent centers produced by a given exposure dose depends on the size of the glass element used in the dosimeter; therefore the smaller the size of this element the higher will be the ultraviolet light intensity and/or photomultiplier amplification required in the fluorimeter.

8.3.1 Materials

Table 8.2 shows the composition of several phosphate glasses and a number of their characteristics. The "High Z" glass was the original radiophotoluminescent glass used in the first U.S. military dosimeter, the DT60/PD, and the "Low Z" glass is one in which the elements barium and potassium were replaced by the lower atomic number elements magnesium and lithium, respectively. The lower limit of detectability with these dosimeters was in the 5-10R range, primarily because of the fluorimeter readings obtained from undosed dosimeters, the so-called "predose" or "background" reading. A major source of the predose signal in these early glasses was unintentional impurities in the glass which functioned as luminescent centers themselves; their visible emission was excited by the ultraviolet light used for reading, and this luminescence was transmitted to the photomultiplier. Careful purification of the raw materials used in preparing the glass subsequently reduced this contribution to the predose signal very markedly, but, of course, did not eliminate it completely. Moreover, since many of the contaminating impurities had a deleterious or "poisoning" effect on the luminescence of the radiation-induced silver centers, their elimination allowed the Ag^0 and Ag^{+2} centers to luminesce more efficiently, thus significantly increasing the dosimeter sensitivity, as may be seen from Table 8.2 (Yokota and Nakajima, 1965).

It is appropriate to mention other sources of the predose signal, some of which relate to the glass and others to the fluorimeter. In addition, of course, there are totally avoidable sources of predose, such as contamination of the glass surface by careless handling or by adherence of luminescent fluids into which the glass may have been immersed, such as body fluids in "in vivo" dosimetry; these contaminants must be avoided, and where their presence is likely, the glass must be carefully cleaned. More subtle causes of predose signal relate to the dosimeter design, and to the luminescence of Ag^+ itself. In the simplest fluorimeter, reliance is placed on an optical filter to exclude the visible light which is also emitted by the ultraviolet lamp, and a second "sharp cut-off"

TABLE 8.2 COMPOSITION AND PROPERTIES OF SOME RADIOPHOTOLUMINESCENT GLASSES.

Source	Name	Composition(% by weight)						Predose (R gamma exposure equivalent) ²	Maximum energy dependence ³	Relative Sensitivity (°C)	Dose Range
		Ag	Al	Li	P	O	Others				
Bausch & Lomb Rochester, N.Y.	High Z ¹	4.3	4.7		28.4	44.1	10.8Ba, 7.7K	10 - 40	30 - 32	1	10R-40R
	Low Z	4.3	4.7	1.9	33.7	52.3	3.1 Mg	10 - 40	7 - 10	1	10R-40R
Toshiba Tokyo, Japan	FD-1	3.7	4.6	3.6	33.5	53.7	0.85 B	0.1-0.3	7 - 10	2.2	40mR - 3000R
	FD-3	3.3	5.1	3.6	34.5	53.5		0.1-0.3	7	2.2	
FD-5		0.52	6.1		33.1	51.3	8.9 Na	0.1	3.5	3.2	"
	FD-7	0.17	6.1		31.5	51.2	11.0 Na	0.05-0.1	2.8	3.6	"
CEC, Montrouge, France	Type 1	2.9	3.5	2.5	33.6	52.3	4.7 Na, 0.5Be	0.6	6.5 - 7.3	1.9	"
	Type 31	2.4	3.5	0.7	34.1	52.8	4.8 Na, 1.7Be				"
Jenaer Glas - werke Schott & Gen., Mainz, Germany	RPL-I	4.6	3.1	4.45	34.1	52.8	0.9 Mg	0.3	7.5 - 11	2.1	"
	RPL-II	3.67	4.67	3.47	33.5	57.7	0.9 B	0.3			"
	RPL-III	0.62	0.51	7.34	34.7	55.85	0.95 B	0.5	4 - 5	3.1	"
	RPL-V	0.58	6.1		32.3	50.2	6.45K, 4.4Na	0.3			"
	9929	0.4	6.1		33.2	51.5	8.86 Na	0.25	3.5 - 4.7	2.4	"

¹ No longer in production² Depends on optical system of the reader, precleaning of the glass (removal of fluorescent surface contamination), and time elapsed since manufacturing or thermal annealing (measurable dose build-up occurs in low-predose glasses during extended storage)³ The lower value is usually the measured and the higher value the calculated ratio between maximum sensitivity at 30-40 keV and maximum sensitivity at 1.25 MeV.

(orange) optical filter in front of the photomultiplier to reduce even further this lamp-generated light (see Figure 8.1). Since both filters are rather broad-band in their transmissions, perfect exclusion of this spurious visible signal cannot be obtained, and this contributes to the predose. Moreover, the orange filter cannot avoid being exposed to scattered ultraviolet light from the lamp; hence if the filter itself is even feebly luminescent, its emitted light will be seen by the photomultiplier and contribute a predose signal. More elaborate readers employing monochromators instead of, or in addition to, the filters can essentially eliminate these sources of luminescence background.

An unavoidable source of predose signal is the Ag^+ itself. The long-wavelength "tail" of the Ag^+ absorption extends into the near-ultraviolet, and the long-wavelength tail of the blueish Ag^+ luminescence extends into the spectral window transmitted by the orange filter. Thus with respect to the predose the Ag^+ acts like one of the contaminants mentioned above, but unlike these contaminants it is an essential ingredient which cannot be removed. The most advanced reading system discriminates against this Ag^+ luminescence by taking advantage of the observation that the "afterglow" or "luminescence lifetime" of the Ag^+ emission is shorter than that of the luminescence from the radiation-induced Ag^0 and Ag^{+2} centers (Kastner, et al., 1967). By using a fast pulse of near-ultraviolet laser excitation and "gating" the photomultiplier to see only light with the longer afterglow time, the Ag^+ predose emission (as well as the luminescence of some unremoveable trace impurities and surface contaminants) is excluded. By this means the predose has been lowered to the equivalent of a few milliR.*

8.3.2 Energy Dependence and Directional Dependence

The energy dependence of the glass is controlled by its composition which determines its radiation-absorption characteristics. As Table 8.2 shows, the original "High Z" glass contained the high atomic number elements barium and potassium, which gave rise to considerable energy dependence. The replacement of these heavy elements in later host glass compositions reduced the energy dependence considerably, but did not eliminate it. Moreover, the high atomic number element silver is essential to the functioning of the glass and cannot be replaced.

* Another technique based on actual decay time measurements of the radiophotoluminescence, rather than on the measurement of intensity using the above gating technique, claims to be able to detect as little as 0.2mR (Barthe et al., 1975). There is no theoretical explanation for the observed correlation of decay time with dose found by these investigators.

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In order to achieve air-equivalent or tissue-equivalent energy response the glass dosimeter must, therefore, be provided with filters ("shields") to cut down the excess low-energy radiation response due to photoelectric absorption by its high atomic number constituents. Solid filters of lead, cadmium, tin, or combinations of these are used to cut out this excess response. It is impossible by shielding to obtain a uniformly flat response down to the lowest energies of interest; the filter thickness required to reduce the response in the mid-range of energies necessarily cuts out too much of the response at the lower energies (Figure 8.13). Some of this low-energy loss can be avoided by using perforated filters which allow a predetermined fraction of the low energy radiation to reach

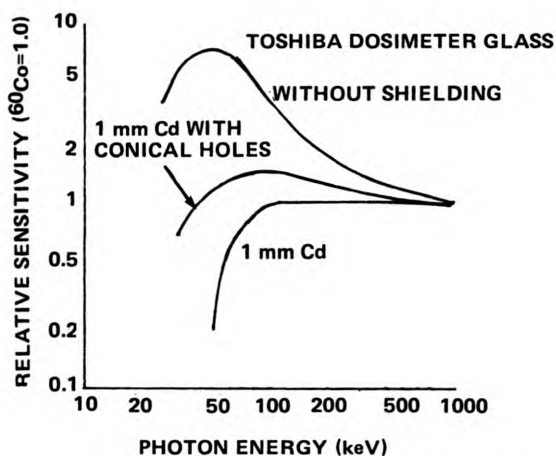


FIGURE 8.13 PHOTON ENERGY DEPENDENCE OF RESPONSE OF TOSHIBA DOSIMETER GLASS BLOCKS WITHOUT SHIELDING, WITH A SIMPLE 1 mm Cd SHIELD, AND WITH A PERFORATED Cd FILTER. (AFTER PIESCH, 1972).

the glass unfiltered (Klick, 1956). The configuration of the perforations and the percentage of the area they uncover depends on the shape of the glass, its composition, and the thickness of the heavy metal filter (Figure 8.14). It also depends on the degree of departure from flat response one is willing to tolerate in the intermediate energy range, since the perforations increase the response at these energies. The design of perforated filters is generally done empirically. Despite the use of perforations, only a portion of the low-energy response can be recaptured, and the higher atomic number glass dosimeters therefore lose useful sensitivity below approximately 40 keV. However, by adding a second glass dosimeter element provided with appropriate filters designed expressly for the low energy part of the spectrum, the entire range of 15 keV to 1.25 Mev can be covered with energy-independence by the dosimeter pair (Figure 8.15).

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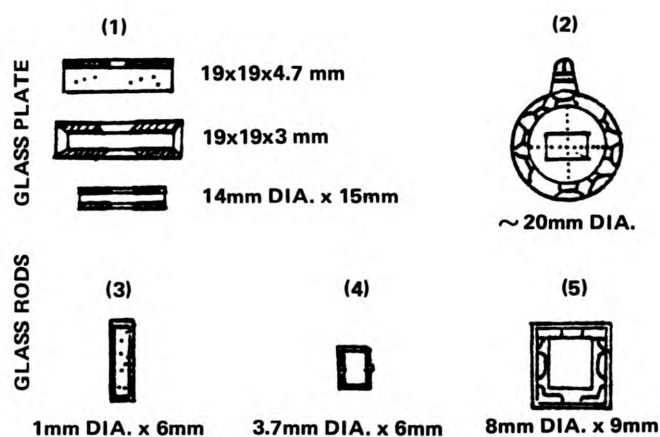


FIGURE 8.14 VARIOUS X-RAY ENERGY-COMPENSATING DOSIMETER CAPSULES FOR PHOSPHATE GLASSES OF VARIOUS SIZES (MAUSHART AND PIESCH, 1966).

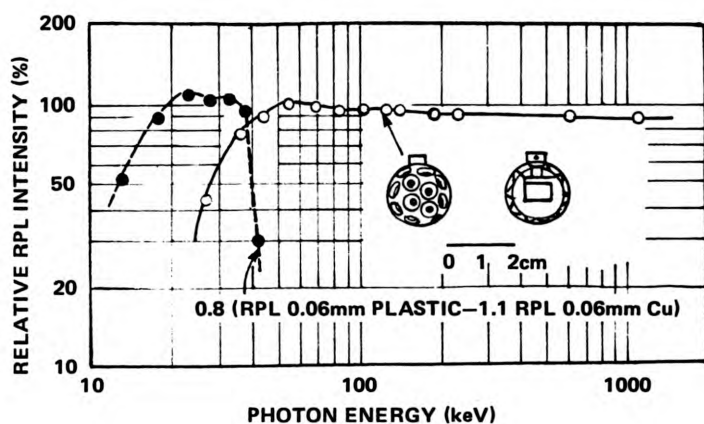


FIGURE 8.15 Photon energy dependence of response of Toshiba FD-3 glass in spherical tin capsule with conical holes and additional set of glasses covered with copper and plastic to facilitate dose determinations at low photon energies. (After Maushart and Piesch, 1965).

For simple geometrical reasons, the use of filters to smooth out the energy dependence generally introduces a directional dependence of response for lower energy radiations. The effective thickness of a flat metal shield applied to the plane surface of a glass block will obviously vary with the angle of incidence of the radiation. The least directional arrangement would occur with a glass sphere surrounded with a spherical shield. An approximation to this is the dosimeter configuration shown in Figures 8.14 and 8.15 in which a cube of glass is enclosed in a perforated metal shield that can be disassembled for removal of the glass for reading. With this

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arrangement, directional dependence is reduced to the order of 10 percent, a factor of 3 to 4 improvement over flat shielding. In general, directional dependence is not a major handicap for many purposes. Conversely, with special readers one can make use of the directional dependence to determine the direction of radiation, where this information is needed.

By the same token, the energy-dependence of radiophotoluminescent glasses is not always a disadvantage. It can be turned to advantage if one wishes to obtain a rough determination of the spectral composition or "quality" of the radiation. For this purpose one can use either separate pieces of glass provided with filters of different radiation-absorption characteristics, or one can use a single piece of glass with different areas covered by different filters. In the latter case a special fluorimeter is employed which can separately measure the radiophotoluminescence from the unfiltered and various filtered portions. Knowing the energy-dependence of the glass and the absorption characteristics of the several filters, the approximate spectral composition of the ionizing radiation can be determined.

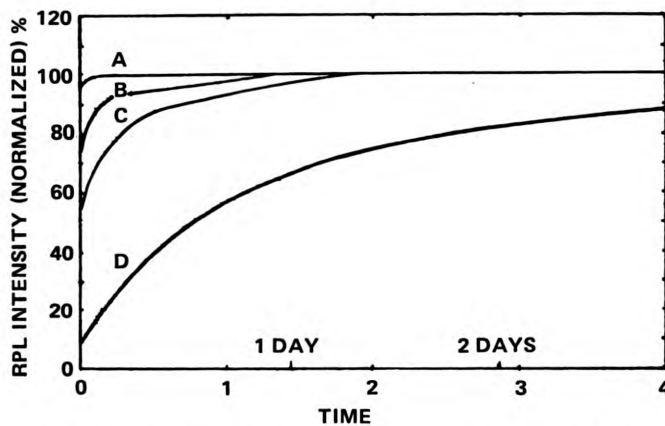


FIGURE 8.16 BUILDUP OF RADIOPHOTOLUMINESCENCE AT 25° C IN Li-Al-METAPHOSPHATE GLASS CONTAINING DIFFERENT PERCENTAGES BY WEIGHT OF AgPO_3 (A, 9-12 WT.%; B, 6 WT.%; C, 4 WT.%; D, 1 WT.%) AFTER A γ -RAY EXPOSURE OF $2 \times 10^3 \text{ R}$ (YOKOTA AND IMAGAWA, 1967).

8.3.3 Dynamics of Radiophotoluminescence Response: Growth, Fading, and Erasure

It has been found that all the radiation-induced photoluminescent centers in the phosphate glass are not formed instantly, but there is a time lapse during which radiophotoluminescent centers continue to grow. This time lapse depends on the composition of the host glass, the Ag^+ content (Figure 8.16) and also on the temperature (Figure 8.17).

SOLID STATE LUMINESCENCE DOSIMETRY

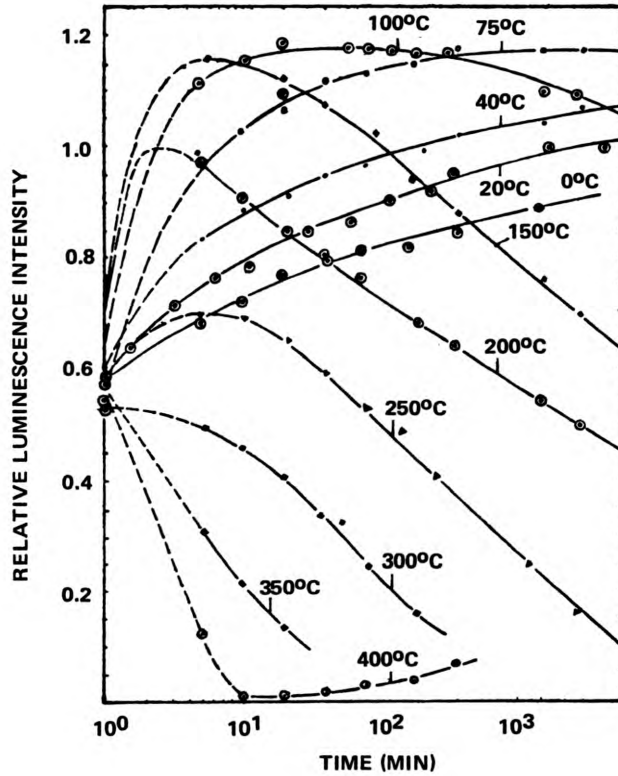


FIGURE 8.17 Radiophotoluminescence intensity of Toshiba FD-1 glass stored at different temperatures, as a function of storage time after receiving a γ -ray exposure in the range 10^{-2} $\sim 10^{-3}$ R delivered within a few seconds (Becker, 1973).

The growth of luminescence centers requires a long time if radiation exposure and storage occur at low temperature; it occurs in the order of several hours in a room temperature environment, and it can take place still faster at a slightly higher temperature. It is believed that the growth process takes place because many of the radiation-induced holes remain trapped at phosphate ions $(\text{PO}_4)^{-3}$. The release of these holes and their migration to Ag^+ ions to form one of the radiophotoluminescent species, Ag^{+2} , are both thermally activated processes; this accounts for both the growth and its temperature-dependence. The same may be true, of course, for the electrons liberated from the phosphate ions in the initial act of absorption of the high-energy radiation; these electrons may be initially trapped at a variety of defect centers in the glass, before they are thermally expelled and migrate to Ag^+ ions and convert them to the other radiophotoluminescent species, Ag^0 . The migration distance of holes and electrons to Ag^+ ions is obviously less if the concentration of silver in the glass is large. This is why the growth rate of centers is more rapid in glasses made with higher silver content.

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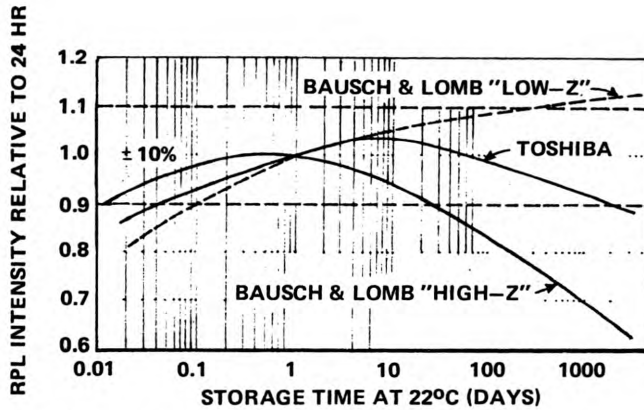


FIGURE 8.18(A) Build-up and fading of radiophotoluminescence glasses (Bausch & Lomb high Z and low Z, Toshiba FD-1) at room temperature (22°C) during ten years of storage, normalized to the radiophotoluminescence one day after exposure. (After Cheka 1968, and unpublished).

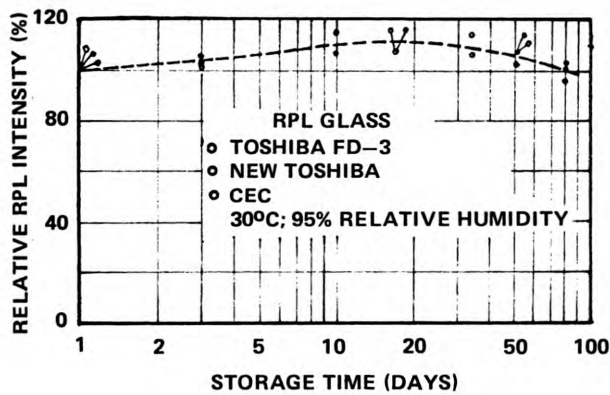


FIGURE 8.18(B) Radiophotoluminescence in three modern dosimeter glasses as a function of storage time at 30°C and 95% relative humidity, normalized to the response one day after exposure.

After radiophotoluminescent centers are created, their stability on storage in the dark is extremely good (Figure 8.18), but it is nevertheless also temperature-dependent. At normal temperatures there is little decrease in dose indication over a period of years. At higher temperatures the radiophotoluminescence reaction can be reversed, thermal energy expelling electrons from Ag^0 and holes from Ag^{+2} to convert them both to Ag^+ and return the glass to its initial condition. This thermal destruction of centers and "regeneration" of the glass proceeds faster at higher temperatures (see Figure 8.17), making it possible to erase the dose

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indication completely and return the glass to its virgin state by an appropriate heating, if it is economically useful to do so. (Obviously, glass elements that are held in place by adhesives, have optical coatings applied to some of their surfaces, or are cemented into plastic holders cannot survive the necessary thermal treatment.) After several such regenerations, however, the glass may suffer noticeable changes in its predose and sensitivity, so that recalibration of its properties is necessary.

The radiophotoluminescent centers are not only stable in the dark at normal operating temperatures, but they are also stable under exposure to the reading light at the ultraviolet light levels normally used. However, prolonged exposure to very intense ultraviolet or visible light will eventually bring about some destruction of the centers, analogous to thermal bleaching, so that unnecessary light exposure should be minimized. This is easily accomplished.

There is no evidence that storage of unirradiated glass for long times at normal temperatures in any way affects its subsequent sensitivity to radiation; in fact, such evidence as does exist indicates that there is no sensitivity change over many years. However, humidity and temperature cycles can condense moisture on the glass surfaces which can dissolve some of the glass constituents preferentially. The dissolved material can then be redeposited as crystalline products on the dosimeter surfaces when environmental conditions change. Both the optical and luminescence properties of the glass are superficially changed by this process, which can strongly affect the predose in particular. Hence over long time intervals, moisture effects must be guarded against.

8.3.4 Reference Standards

As we have pointed out, all luminescence dosimetry methods are secondary or relative methods, hence a reference standard must be used to convert the light output into dose. In radiophotoluminescence dosimetry the output of the photodetector depends on several factors: the size of the glass dosimeter element, its geometry and the geometry of the optical cavity into which it is inserted for reading, the spectrum and intensity of the ultraviolet lamp used for excitation, the spectral response curve and the gain of the photomultiplier employed, the electrical circuit design, and the temperature. In any given dosimeter/reader system, all of these factors remain constant except the ultraviolet light output, the temperature, and the effective photomultiplier gain. The photomultiplier amplification is generally varied to compensate for variations in the other two conditions, using a luminescent reference standard for calibration.

The preferred calibration standard, of course, is an actual dosimeter exposed to a known dose. All features of the luminescence of such a standard would exactly match those of an "unknown"

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dosimeter whose dose one wished to evaluate - spectral distribution of absorption and emission, afterglow time, geometry, temperature dependence, etc. However, because of the slow fading of such a standard over a long period of use, either a fading correction would have to be made or fresh standards would have to be prepared from time to time. For use in the field, a more convenient calibration standard is an artificial one - a radiation-insensitive glass whose luminescence response is due to a chemically incorporated activator. This artificial "working standard," of course, must first be calibrated against an actual dosimeter exposed to a known dose.

Ideally, the working standard should stimulate a real dosimeter as closely as possible. Size, geometrical shape, surface finish, and other gross features are easily duplicated. But the luminescence properties of chemical activators, such as Mn^{+2} and Sm^{+3} which are normally used, do not exactly match those of the radiophotoluminescent centers Ag^0 and Ag^{+2} , although Mn^{+2} - and Sm^{+3} -activated glasses are excited by the same general range of near ultraviolet light and emit orange-red light. In particular the afterglow times of the radiophotoluminescent silver centers differ markedly from those of the above mentioned chemical activators. This has important consequences if readers with different light-source wave-forms or photodetector electronics are employed; the dosage-equivalence of the artificial standard will not be the same for the different readers because of the different frequency dependences of these instruments. Standards of the above type are valid only for readers of the same optical and circuit characteristics for which they have been calibrated.

8.3.5 Dosimeter and Reader Design

The design of radiophotoluminescent dosimeters and of the fluorimeters to read them depends on the application (accident monitoring, health physics personnel monitoring, or in-vivo radiation dosimetry), on the nature of the radiations to be measured, and on whether the system is intended for laboratory or field use. The context in which the system is to be used will determine the dose range of interest, the accuracy and precision of the system, the size of the glass element, the shielding configuration, and the required optical and electrical parameters of the reader.

A military casualty dosimeter, the Navy DT-60/PD Radiac Detector, is among some typical designs shown in Figure 8.19. It consists of a block of silver-activated phosphate glass cemented into the "base" half of a plastic locket, about the size of a pocket watch, which can be hermetically sealed by tightly screwing on the locket cover. Perforated lead shields are built into the base and cover portions of the locket to reduce the energy dependence by the means previously discussed. The optical design is simple; all but two of the surfaces of the glass block are painted with optical black paint - the bare surfaces being one of the larger faces, which is the one

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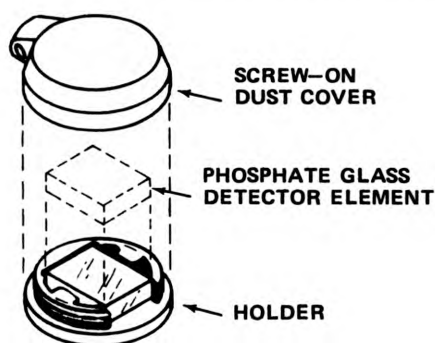


FIGURE 8.19 DT-60/PD PHOSPHATE GLASS DOSIMETER. ENERGY-COMPENSATING PERFORATED METAL SHIELDS ARE INCORPORATED IN THE HOLDER AND COVER.

facing the ultraviolet source in the reader, and one of the smaller surfaces, at right angles to the first, which faces the photo-multiplier tube. Although there is no requirement that the uventrance and luminescence-exit faces be optically polished, they have smooth finishes of the type obtained from molding the glass or fracturing it. Ground surfaces have been found to be more easily attacked over long periods of time by the temperature-humidity cycles previously referred to. The adherence of the optical coating must not change with time, the plastic material comprising the locket and the adhesive used to cement the glass and the lead shields in place must not have any luminescent components. Corrosion of the lead shields - which can produce luminescent deposits - must be prevented. For this reason and for avoidance of glass attack, the closure and gasketing design must be of high quality. The primary features of this system are: a lower detection limit of 5-10R; large range (5-1000R); long life under a wide range of very harsh conditions (extreme temperatures, humidities, contaminations, and shock); rapidity and ease of reading; repeatability of reading with retention of dose indication; and low cost.

The design of the military fluorimeter, the Navy CP-95/PD, is correspondingly kept optically and electrically simple and rugged. The optical cavity into which the glass-bearing portion of the dosimeter locket is inserted contains no mirrors or other optically degradable surfaces: it is a black chamber with two open apertures at right angles to each other, the first being covered by the ultraviolet-transmitting filter and the other by the orange transmitting filter. Various types of exciting lamps can be used, a "blacklight" fluorescent lamp which achieves high output in the near ultraviolet from a uv-emitted phosphor, or, in some reader designs a high pressure mercury lamp. Most of the readers have been designed for line operation, but a battery powered reader using a pulsed flashlamp for excitation has also been developed.

Dosimeters and readers for laboratory operation in clinical application or for large scale personnel monitoring can afford to be

more elaborate in optical and circuit design. The use of computer control of the reader can automate the reading and recording of dosimeters, making feasible automatic reading systems for dealing with large populations of dosimeters. The full degree of this automation, including the measurement and recording of radiation quality information, is embodied in a system developed in Germany which scans different areas of dosimeters that have multiple filters of different absorption coefficients for high energy radiation, computes the effective energy of the radiation, and records the radiation dose, all in one operation (Rober et al., 1980).

8.4 THERMOLUMINESCENCE DOSIMETRY SYSTEMS

8.4.1 General Properties of Thermoluminescent Materials and Dosimeters

Despite a continuing twenty year search for useful phosphors, the materials most widely employed in thermoluminescence dosimetry are comparatively few in number: lithium fluoride, calcium fluoride, and lithium borate (Table 8.3). Other phosphors also shown in the table are useful for more specialized purposes. LiF was the first thermoluminescent phosphor commercially available in ample quantities and a variety of forms. The other major thermoluminescent materials discussed below were available only as laboratory research items in the early days of thermoluminescence dosimetry, but they have since become commercially available as well, and their use in thermoluminescence dosimetry has expanded considerably. None of these solids totally satisfies all the ideal requirements for a thermoluminescent phosphor, even the three most generally useful materials representing compromises in properties. For example, the most sensitive phosphor is $\text{CaSO}_4\text{:Mn}$, but its poor fading characteristics militate against its use except when the time interval between exposure and reading can be determined. $\text{CaF}_2\text{:Mn}$ is probably the next most sensitive material, but (as with CaSO_4) the relatively high atomic number of its constituents gives it an energy-dependent response which is undesirable for many applications. LiF:Mg:Ti is much less energy-dependent, but it has a complex trap distribution which requires rather bothersome pre- and post-irradiation heat treatment. $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$ is even less energy-dependent than LiF, but its radiation sensitivity is low and it is sensitive to humidity.

In the continuing search for better TLD materials there are two guiding principles which can be relied on: the host material should be of low atomic number to minimize energy-dependence, and well-known activator ions like Mn^{+2} , Ag^{+1} , and rare-earths should be incorporated to impart luminescent properties (with preference for those activators which emit in the blue, for reasons we have already

TABLE 8.3 CHARACTERISTICS OF COMMERCIALY AVAILABLE THERMOLUMINESCENT PHOSPHORS.

Phosphor characteristics	LiF:Mg:Ti	Li ₂ B ₄ O ₇ :Mn	CaF ₂ :Nat	CaF ₂ :Mn	CaF ₂ :Dy	CaSO ₄ :Mn	CaSO ₄ :Tm	CaSO ₄ :Dy	BeO
Photon effective atomic number	8.2	7.4	16.3	16.3	16.3	15.3	15.3	15.3	7.1
Ratio of TL response: 30keV, ⁶⁰ Co	1.3	0.9	13-15	13-15	13-15	10-12	10-12	10-12	1.0
Main glow peak temperature(°C)	190 - 210	200-220	200,275	260	200,240	110	220,250	220,250	180,220
Spectral emission peaks (nm)	400	600	380	500	480,577	500	452	478,571	330
Fading of dosimetry peak at normal ambient temperature	5% in 3-12 months	5% in 3 months	3% in 9 months	10% in first 24h	25% in 4 wks(10% in first 24 h)	35% in 24h	6% in 6 months	6% in 6 months	15% in 5 months
Useful range of absorbed dose(Gy)	5x10 ⁻⁵ - 10 ³	10 ⁻⁴ -10 ⁴	10 ⁻⁵ -10 ²	10 ⁻⁵ - 2x10 ³	10 ⁻⁶ -10 ³	10 ⁻⁷ -10 ²	10 ⁻⁶ -10 ²	10 ⁻⁶ -10 ³	10 ⁻⁴ -10 ³

discussed). However, there are no corresponding principles which tell us how to build in only a single type of trap with optimum trap depth, E , and frequency factor, s , in any given host solid. For this reason the trapping properties of phosphors are arrived at empirically and usually are far from ideal. In addition to the currently most important materials, some of the newer promising materials will also be briefly described in the next section.

Thermoluminescent phosphors are used in a variety of physical forms and in different configurations: single crystals, extruded or sintered bars or rods, loose powders, coated tapes, and impregnations into plastic (Teflon) sheets. The phosphors can also be enclosed in glass bulbs like miniature radio tubes or light bulbs, in which the powdered material is coated onto a heating filament, or bars of the phosphor are clamped in tight contact with a heating element (Schulman et al., 1960). Handling procedure and reading instrumentation vary depending on these different configurations. For example, loose powder is subject to rubbing of the grains together in transfer from the packages in which they are exposed to the metal tray or planchet in which they are heated during the reading process; this can give rise to the spurious "tribothermoluminescence" mentioned earlier. In all cases, intimate thermal contact between the phosphor and the heating element is required for fast and reproducible readings. To deal with both problems, flowing hot nitrogen or argon is used in some reading instruments, the hot gas making good contact with the phosphor grains and nitrogen suppressing the tribothermoluminescence or spurious luminescence by some mechanism not yet understood. For another example phosphors impregnated into Teflon cannot be heated substantially above 300 °C. For phosphors such as LiF which require a 400 °C heat treatment to restore their original glow curves after use, adequate regeneration by heat-treatment of the Teflon dosimeters may be impossible. As a final example, intimacy of contact with the heating element, and protection from mechanical disturbance and chemical contaminants is best achieved in the sealed-bulb type of dosimeter; this type however, is less versatile (because of size, imposition of the glass between the phosphor and the radiation, and the proximity of high atomic number filament metals) than loose powders, sintered phosphors, or plastic-impregnated tapes, which can be made quite small and are adaptable to many applications, including in-vivo insertion.

It is not feasible to discuss in detail all the phosphors that have achieved some measure of use in thermoluminescence dosimetry, nor the differences in properties between different manufacturers' products or different forms of a given phosphor. Table 8.3 gives an overview of the principal current materials (LiF, $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$ and $\text{CaF}_2\text{:Mn}$) which will be discussed in greater detail in the text. General comparison of the glow curves, emission spectra and energy dependence of these thermoluminescent phosphors is given in Figures 8.20, 8.21 and 8.22 respectively.

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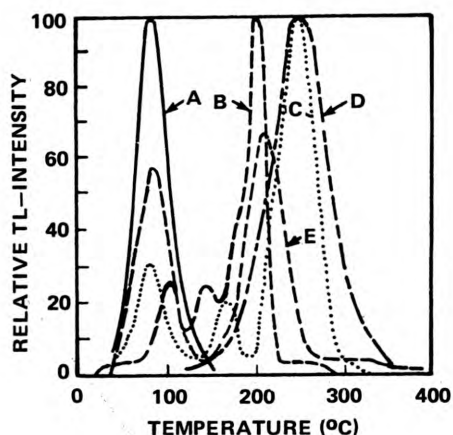


FIGURE 8.20 GLOW CURVES OF SOME THERMOLUMINESCENT PHOSPHORS.
(A) $\text{CaSO}_4\text{:Mn}$, (B) LiF:Mg:Ti , (C) NATURAL CaF_2 , (D) $\text{CaF}_2\text{:Mn}$,
(E) $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$.

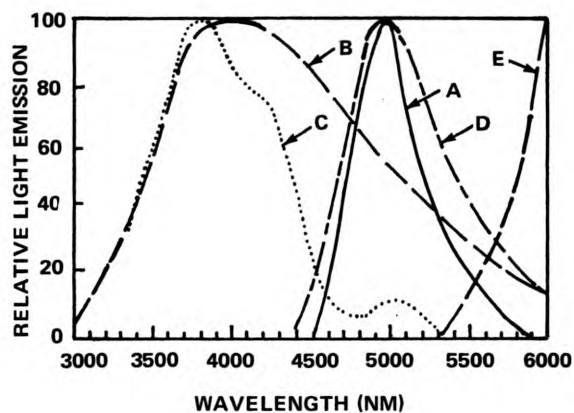


FIGURE 8.21 EMISSION SPECTRA OF SOME THERMOLUMINESCENT PHOSPHORS.
(A) $\text{CaSO}_4\text{:Mn}$, (B) LiF:Mg:Ti , (C) NATURAL CaF_2 (260°C PEAK), (D)
 $\text{CaF}_2\text{:Mn}$, (E) $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$.

8.4.2 Specific Thermoluminescent Phosphors

Lithium Fluoride

Pure LiF is not useful as a thermoluminescent phosphor. The thermoluminescent dosimetry properties of LiF are attributed to the presence of magnesium and titanium impurities, and perhaps to still other impurities such as aluminum and sodium. The precise constitutions of the luminescent and trapping centers in LiF:Mg:Ti

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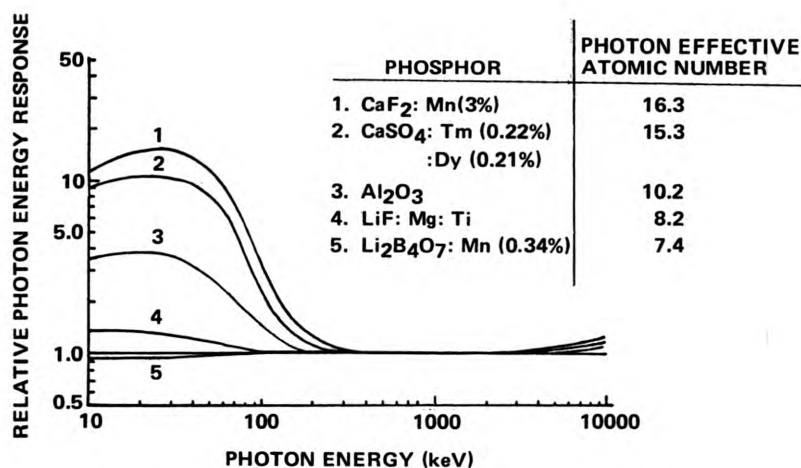


FIGURE 8.22 THEORETICAL PHOTON ENERGY RESPONSES OF A NUMBER OF THERMOLUMINESCENT PHOSPHORS RELATIVE TO THAT OF AIR (1.00).

are still not established, although it seems clear that Mg^{+2} , Mg^{+2} -vacancy complexes and their aggregation into "dimers" or higher clusters are involved. Since the formation and dissociation of these complexes are temperature-dependent processes, even with a specified chemical composition the nature and distribution of the several possible types of trapping center depend on the thermal history of the sample (Figure 8.23 and 8.24). The process of reading

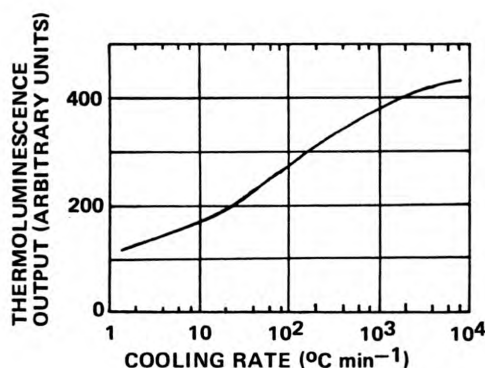


FIGURE 8.23 THERMOLUMINESCENCE SENSITIVITY OF LiF:Mg:Ti AS A FUNCTION OF COOLING RATE FROM 400°C .

a thermoluminescent dosimeter is obviously, in itself, a thermal treatment, and as such it affects the trapping configurations. Moreover, it has been found that doses of ionizing radiation of the order of several thousands of rads can create new traps or otherwise affect the luminescent properties of LiF:Mg:Ti to increase its sensitivity, thus producing a so called "superlinear" response to

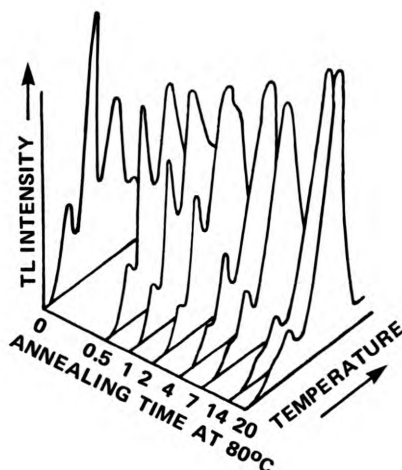


FIGURE 8.24 DYNAMICS IN THE GLOW-CURVE STRUCTURE OF LiF:Mg:Ti AS A FUNCTION OF POST-READ-OUT ANNEALING TIME AT 80°C. ANNEALING TIME IN HOURS.

dose and altering the subsequent sensitivity of the dosimeter (Figure 8.25). In order to counteract all these effects, LiF dosimeters must undergo a pre-exposure heat treatment (called an "annealing"). A "standard" annealing cycle is a 1-hour heating at 400°C followed by a 24-hour heating to 80°C. A faster annealing cycle has also been recommended, 5 minutes at 400°C, followed by a 10 minute post-dose heating at 100°C, but the glow curve obtained in this way is not identical with that obtained after the "standard" annealing. After exposure to very high doses (10^5 rads), thermal annealing will no longer restore LiF to its initial sensitivity and glow curve.

Because of the low-temperature components in the glow curve of LiF:Mg:Ti, the phosphor must also be given a post-dose heat treatment, generally for 10 minutes at 100°C. Various preparative methods produce LiF phosphors with substantially different properties, including different requirements for pre-exposure and post-dose annealing. In all cases, therefore, one must scrupulously adhere to the operating procedures specified for the type of LiF preparation being used if consistent and correct results are to be obtained.

It may seem remarkable that, despite all these complications, LiF achieved such early and sustained popularity. This can be understood, however, by considering the many advantages it offers as a dosimetric material: sensitivity permitting measurement of doses from the milliR range to well beyond the kiloR range, dose-rate-independent response at least as high as 10^{11} rads/sec, low energy dependence (Figure 8.22), and good dose storage properties (Figure 8.26).

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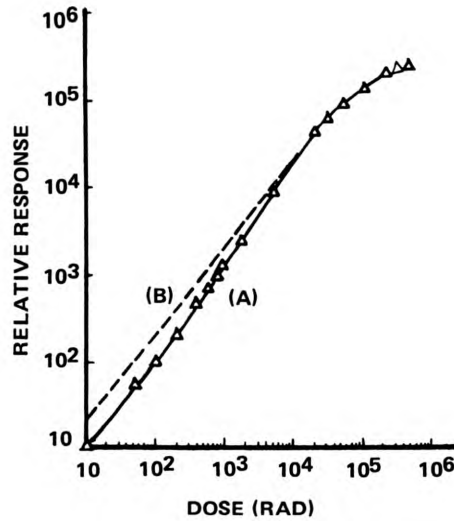


FIGURE 8.25 DOSE CHARACTERISTIC OF LiF. (A) ORIGINAL MATERIAL
(B) PRE-IRRADIATED (10^4 RAD) MATERIAL.

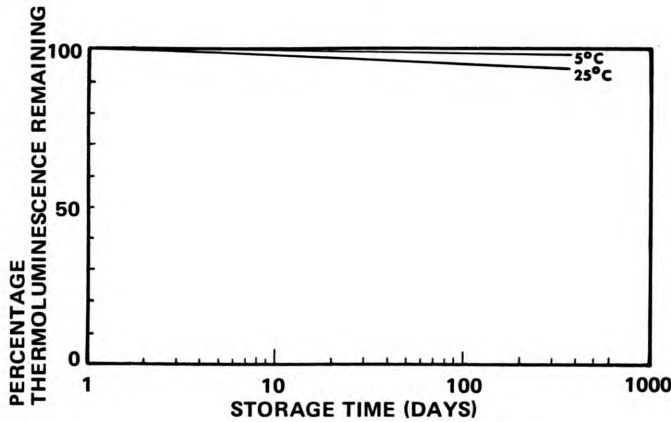


FIGURE 8.26 THERMAL FADING OF LiF:Mg:Ti FOR STORAGE TEMPERATURE OF 5° AND 25° AFTER 20 MINUTE PRE-READ ANNEAL AT 100°C .

LiF:Mg:Ti is sensitive to thermal neutrons (neutrons of energies less than 0.5 electron volts) because of the high capture cross section of ^6Li and the resulting reaction: $^6\text{Li} + ^0_1\text{n} \rightarrow ^3\text{H} + \alpha$, both products producing ionization in the solid. Isotopic purification of the natural LiF to give nearly 100 percent ^7LiF or 100 percent ^6LiF can reduce or augment this response respectively. A comparison of the thermal neutron response of these isotopically different phosphors is given in Table 8.4 which includes the response of $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$ with a natural $^6\text{Li}/^7\text{Li}$ ratio.

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TABLE 8.4 THERMAL NEUTRON RESPONSE OF Li-BEARING PHOSPHORS.

PHOSPHOR	THERMAL NEUTRON RESPONSE (EQUIVALENT Co DOSE IN R PER 10^{10} NEUTRONS/cm ²)
$^6\text{LiF:Mg:Ti}$	870-2190
$^7\text{Li:Mg:Ti}$	0.7-2.5
LiF:Mg:Ti (Natural)	85-535
$\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$ (Natural)	230-500

Lithium Borate

Manganese-activated $\text{Li}_2\text{B}_4\text{O}_7$ was introduced in 1965 (Schulman et al., 1965) as a low-atomic-number alternative to LiF, its energy dependence being even less than that of LiF (Table 8.3 and Figure 8.22). Its thermoluminescence emission, however, lying in the red end of the spectrum (Figure 8.21), makes it considerably less sensitive than LiF, although despite this lower sensitivity its dose range extends linearly from 10 milliRad to 300 Rad. The glow curve of $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$ is relatively simple, consisting of only two glow peaks, a composite peak at approximately 55-90°C and a single peak at 200°C (Figure 8.20). The lower-temperature peak, of course, fades rapidly, and only the 200°C peak is used for dosimetry. The erasure of the low-temperature peak can be accelerated by a short low-temperature post-dose heating, as is done with LiF. A disadvantage of $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$ is its hygroscopic nature which causes it to deteriorate if exposed to high humidity. A different formulation of the phosphor, containing 0.5 SiO₂, has reportedly overcome this handicap. Unlike LiF, $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$ does not require a complex pre-exposure annealing cycle to retain its original thermoluminescence characteristics. Samples that have been exposed to very high doses, however, require heat treatment before they can be used again to register low doses, since the short normal thermoluminescent reading cycle does not completely empty the high-temperature traps. A fifteen minute heating at 300°C suffices to do this. The storage of dose indication is good, room temperature fading being 10 percent in 2 months and 37 percent after 13 months. $\text{Li}_2\text{B}_4\text{O}_7\text{:Mn}$ is now commercially available in several forms.

In the attempt to increase the sensitivity of $\text{Li}_2\text{B}_4\text{O}_7$ by shifting the luminescence toward shorter wavelengths, copper and silver have been substituted as activators to replace the manganese (Takenaga et al., 1977). The resulting phosphor has two glow peaks,

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a low-temperature peak at 110°C and one at 185°C to 205°C ; the emission spectrum has three bands, two in the ultraviolet and one in the green. The sensitivity of the phosphor is about twenty times as high as the Mn-activated salt, its linearity of response extends to about 10^5 Rad, and the storage stability is good in the dark at room temperature (less than 9 percent fading in 60 days). The major drawback of this phosphor is that the traps are emptied rapidly by exposure to normal light (80 percent fading after a 7 hour exposure to normal ambient fluorescent lighting conditions). This deficiency does not seem to present an operational problem. The $\text{Li}_2\text{B}_4\text{O}_7\text{:Ag;Cu}$ phosphor is not yet commercially available.

Calcium Fluoride

Calcium fluoride occurs as the natural mineral fluorite, some deposits of which contain impurities which render them luminescent and thermoluminescent with rather complex glow curves (Figure 8.20). Different sources of the mineral give samples with different glow curves and emission spectra. By selecting and blending lots of the natural mineral to average out quantitative variations in properties, large supplies of "dosimetry grade" phosphors can be established. The luminescence emission of preferred types of natural CaF_2 is in the blue, and the sensitivity of this material is as much as fifty times greater than that of LiF:Mg:Ti .

A more conventional approach to material supply, of course, is the preparation of a synthetic CaF_2 phosphor activated by one of the activators previously discussed. Manganese-activated CaF_2 was the first of these synthetic fluorites, and it is still one of the most important (Ginther and Kirk, 1957). Its glow curve, (Figure 8.20) is simple in appearance with apparently a single high temperature peak. Research has shown, however, that this broad curve must be composed of several overlapping and non-resolvable glow peaks. The blue-green emission of $\text{CaF}_2\text{:Mn}$ lies in a favorable region of photomultiplier sensitivity, and the phosphor is 3 to 5 times more sensitive than LiF:Mg:Ti . Its large linear dose range from a few milliR to $2 \times 10^5\text{R}$, freedom from superlinearity and other effects that require special heat treatments, its dose-rate independence and its excellent post-dose stability (Figure 8.27) make it the phosphor of choice for many applications. A major drawback of the phosphor is its energy dependence (Figure 8.22); this must be corrected by appropriate filters in the design of the dosimeter. Its small amount of sensitivity to visible light is also taken care of by the opaque dosimeter case. $\text{CaF}_2\text{:Mn}$ is commercially available in several forms and is used in the Navy's DT-526/PD dosimeter, a glass-bulb type with internal heating element (see Section 8.5).

Dysprosium and thulium activated calcium fluoride, $\text{CaF}_2\text{:Dy}$ and $\text{CaF}_2\text{:Tm}$, are also phosphors that have achieved use because of high sensitivities and other properties given in Tables 8.3 and 8.4.

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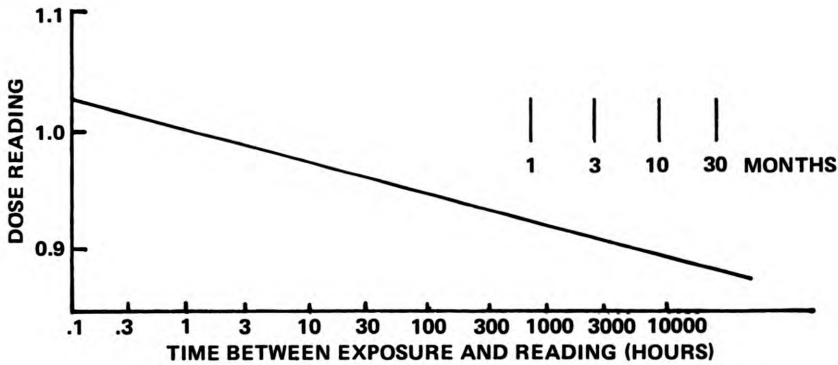


FIGURE 8.27 $\text{CaF}_2\text{:Mn}$ THERMOLUMINESCENCE AS A FUNCTION OF TIME AFTER EXPOSURE. (U. S. NAVY SEALED BULB DT-526/PD DOSIMETER).

Other Thermoluminescent Phosphors

In Table 8.5 we comment briefly on the characteristics of other phosphors that have achieved some degree of interest. Further details and references to the original literature are given in the references following the table.

8.4.3 Types of Thermoluminescent Dosimeters

The simplest form of thermoluminescence dosimeter is a capsule or packet of powdered thermoluminescent phosphor. After radiation exposure, a predetermined volume or weight of the powder is transferred to a shallow metal pan or planchet which is placed into the reader chamber and warmed by contact with a heated block or by some other means (see Sections 8.2.5 and 8.2.6). Alternatives to powdered phosphor are sintered phosphor blocks or rods, phosphor-impregnated polymers, or other similar configurations. The packaging and handling of the active phosphor material must take into account any sensitivity it may possess to ambient light, mechanical handling or friction, and its energy dependence. Light sensitivity is countered by opaque container material and handling in subdued illumination. Correction for energy dependence is made by use of filters or shields, in the same manner previously described for radiophotoluminescent dosimeters. Tribothermoluminescence effects and other spurious light signals can generally be suppressed by carrying out reading procedure in an atmosphere of inert gas. The problem of obtaining good thermal contact between the phosphor and the heating element is an important consideration, and the design of the reader must accomplish this (see Section 8.4.4).

Of particular interest for military personnel monitoring is the sealed bulb dosimeter design, in which a powder phosphor coating is bonded to a filament in a miniature bulb resembling a light bulb, or a bar of the phosphor is tightly clamped to the heating filament

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**TABLE 8.5 PROPERTIES OF SEVERAL THERMOLUMINESCENT
DOSIMETRY PHOSPHORS**

CaSO₄:Mn	Easily prepared with limited laboratory facilities. Simple glow curve, peaking near 100°C*, with consequent rapid fading. Very high sensitivity. Useful for short term monitoring.
CaSO₄:Sm	Two glow peaks, at 100 and 400°C*. Fading of high temperature peak is negligible, but red emission (peaking near 600 nm) limits sensitivity to about 1R.
CaSO₄:Dy CaSO₄:Tm	Three glow peaks at 80, 120, and 220°C*, the last being used for dosimetry. At high exposures a 250°C glow peak appears and dominates the glow curve. CaSO ₄ :Tm emission peaks at 452 nm and CaSO ₄ :Dy emission peaks at 478 and 571 nm. High sensitivity and low fading in both phosphors. Useful for environmental monitoring.
CaF₂:Dy	Complex glow curve with as many as six peaks. Sensitivity good but fading rapid. Useful for environmental monitoring with appropriate procedures taking fading into account.
CaF₂:Tm	Complex glow curve with main peaks at 150 and 250°C*, the latter being supra-linear with dose. Blue emission spectrum with peaks at 450 and 475 nm. Indications that phosphor may be useful for high l. e. t., neutron and mixed field dosimetry.
BeO	Low energy dependence, low sensitivity, low fading. Variable glow curve depending on conditions of preparation. Less sensitive than LiF:Mg:Ti. Powder is toxic.
Al₂O₃	Complex non-linear behavior of glow curve. Less sensitive than LiF:Mg:Ti.
MgB₄O₇:Dy MgB₄O₇:Tm	Recently announced (Prokic, 1980) phosphors. Main glow peak at 210°C*. 5-10 times as sensitive as LiF:Mg:Ti. Low fading, resembling Li ₂ B ₄ O ₇ :Mn, with energy dependence resembling LiF. Simple annealing procedure. Not hygroscopic.

* At normal thermoluminescence heating rates.

without a cement. This is the design used in the Navy Radiac Detector DT-526/PD which can be used to measure gamma and x-rays doses from 5 mR to 5000R. Figure 8.28 shows an exploded view of the dosimeter and its case. The phosphor used is CaF₂:Mn, and its energy-dependence is corrected by a combination lead and tin shield to keep its response constant within about 20 percent over the range of energies 80 keV to 1.3 MeV. The Navy reader for this dosimeter is described in Section 8.4.4.

8.4.4 Thermoluminescence Dosimeter Readers

Multipurpose Readers

Multipurpose thermoluminescence dosimetry readers must have provision for accepting different forms of solid (powder, blocks, tapes, etc.); controlling the heating rate by a linear or other desired schedule of temperature rise; programming the heating cycle

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to include post-dose heating to empty shallow traps if necessary; inert atmosphere for phosphors that exhibit background spurious or tribothermoluminescence; circuitry for measuring either glow-curve peak height or light sum, whichever parameter is more suitable for the phosphor being used; and a post-reading anneal period to completely empty out all the traps to "zero" out the dosimeter for its next use. For materials like LiF where long annealing is necessary to restore the trapping defect concentrations and configurations, separate annealing ovens are needed in order not to tie up the reader. Several commercial designs of multipurpose readers exist.

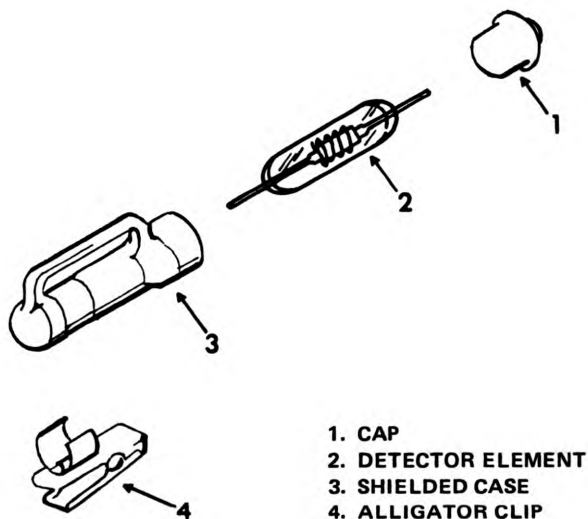


FIGURE 8.28 RADIAC DETECTOR MODEL DT-526/PD, EXPLODED VIEW.

Special Purpose Readers: U.S. Navy Bulb Dosimeter Reader

Single purpose readers dedicated to particular dosimeter configurations and dosimetric tasks can, of course, dispense with some of this versatility and concentrate on other desired features: automatic loading and reading, portability, ruggedness in field use, etc. A particular example of interest is the Navy Radiac Computer Indicator CP-1112/PD designed to read the Navy Radiac Detector DT-526/PD sealed-bulb thermoluminescent dosimeter (Figure 8.28). This is shown in Figure 8.29.

In operation, the dosimeter bulb is removed from its case and inserted into the reader drawer. A 6 amp. current is passed through the bulb, while a strip chart recorder records the glow curve, automatic scale changes being made as necessary to accommodate the magnitude of the dose between 1 mR and 5 kR. At the same time a digital display gives a reading of the dose after 12 seconds. The recorder stops after 15 seconds and the heating cycle is terminated

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at 20 seconds. The reference light source built into the instrument is a $\text{CaF}_2:\text{Mn}$ phosphor chip excited by the radiation from approximately 8 microcuries of carbon-14 absorbed on the surface of the chip. The reader is line operated.

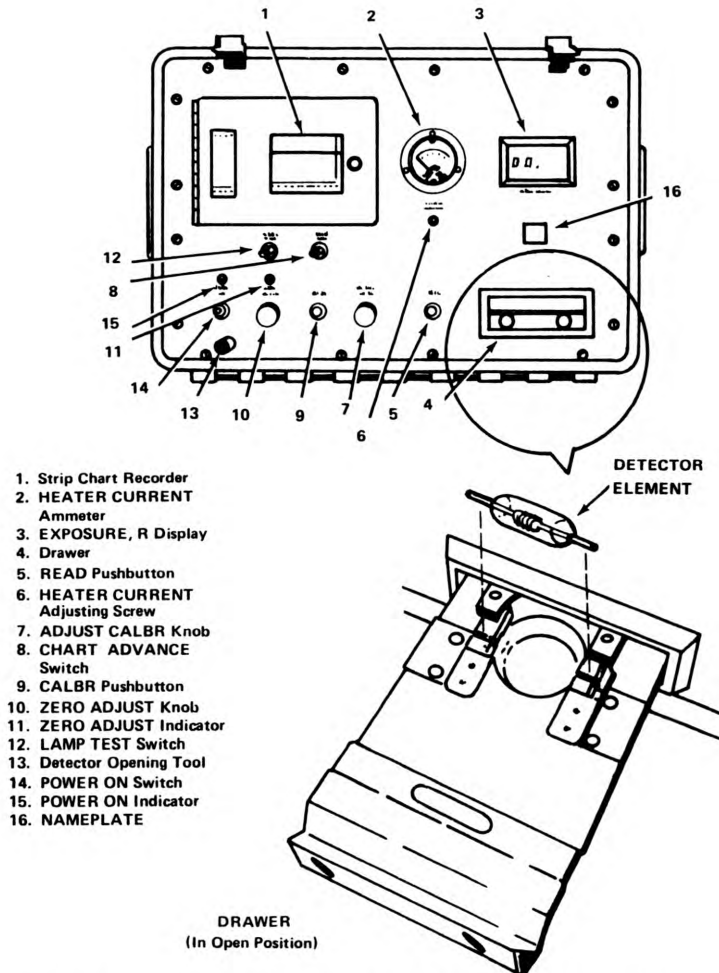


FIGURE 8.29 TLD READER MODEL CP-1112/PD FOR RADIAC DETECTOR MODEL DT-526/PD.

New High-Speed Heating Methods

All thermoluminescent dosimeter-reader designs have one problem in common - the need to insure rapid and controllable heating of the dosimeter element. Rapid heating is desirable for at least two reasons: first, operational speed is necessary if large numbers of dosimeters are to be read quickly, and secondly, higher sensitivity is obtained when the measured quantity is glow-peak height. Controllability is necessary, because the reproducibility

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of the temperature schedule is absolutely vital, particularly if discharge of low-temperature traps is accomplished in the reader, whether the quantity measured is peak height or light-sum. Practically all thermoluminescent dosimetry systems depend on contact heating of the phosphor, i.e., transfer of the heat energy from a heated support to the phosphor. In most cases the support itself is a metal which is ohmically heated either directly (as in the case of a filament in a sealed-bulb type of dosimeter) or indirectly (as in the case of trays or planchets which are placed in contact with an electrically heated block). Other heating means are radio-frequency heating, wherein the dosimeter is placed on a support which absorbs radiofrequency energy; basically, however, this does not change the nature of the heat-transfer problem. Yet another solution has been the use of hot inert gas as a heating agent, which gives intimate contact with all surfaces of the dosimeter rather than restricting heat flow unidirectionally. In general, because of the heat-transfer problems inherent in contact heating, heating rates have been limited to about 10°C per second.

While still employing contact heating, this rate has been exceeded by an order of magnitude by use of infrared radiant heating, coupled with clever dosimeter and reader design. Ordinary infrared sources, of course, also produce visible light, and it is difficult to filter out this light and prevent it from being reflected into the photodetector. This problem is largely solved by depositing a coating of phosphor on a carbon-coated polymer (polyimide) film, and providing the light source with a silicon filter. By this means a dosimeter consisting of a thin coating of a phosphor with a 200°C glow peak can be read out in less than a second (Figure 8.30) (Matsushita Electric Industrial Co., Ltd., Japan).

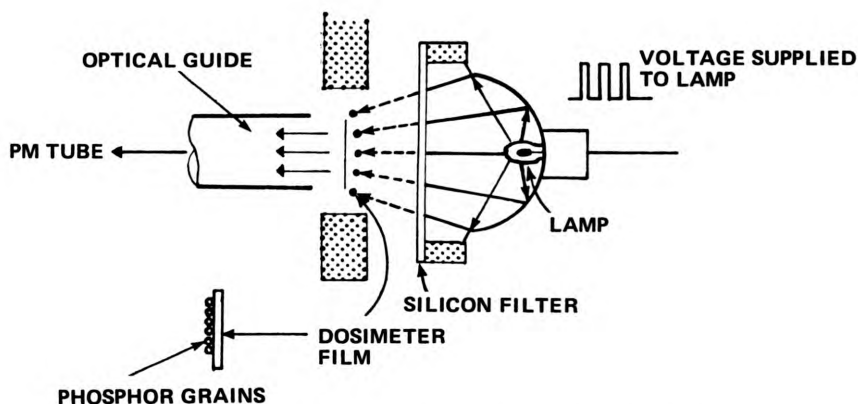


FIGURE 8.30 RADIANT HEATING SYSTEM. A 1 MG SAMPLE OF $\text{CaSO}_4: \text{Tm}$ POWDER CAN BE READ OUT IN LESS THAN ONE SECOND.

The above ingenious arrangement nevertheless depends on contact heating of the phosphor by heat transfer from the carbon-

coated support. In general, the faster one wishes to heat the phosphor by thermal conduction from a support, the higher must be the temperature difference between the support and the phosphor. Even with a high-melting support such as the usual metal planchet or filament, the limit on heating rate occurs when incandescent light from the support becomes sufficiently intense to interfere with the thermoluminescent glow signal. Direct laser heating of the phosphor can be employed in favorable cases, however, avoiding the problem of heat flow across an interface and permitting the attainment of extremely rapid heating rates estimated to be of the order of 10^4 °C per second in recent laboratory work (Braunlich et al., 1981; Gasiot et al., 1982). For this purpose the host phosphor crystal must have a strong absorption band that matches the laser emission. If a pulsed carbon dioxide laser source is used, emitting at $10.6\ \mu$, CaSO_4 phosphors can be successfully employed in this manner because the CaSO_4 crystal has a strong absorption at that wavelength. CaF_2 and LiF , however, have only weak absorption in that region and are not prime candidates for this heating method. However, even these lesser absorbing phosphors can be used in what may be called a "hybrid" heating method. For this purpose thin layers of the phosphors, of the order of about 20 microns, are deposited on a glass microscope cover slide and cemented into place with a high temperature polymer that absorbs 10μ radiation. On illumination of the dosed samples with carbon dioxide laser pulses, heating occurs both by direct laser light absorption (weakly) and by contact transfer from the polymer cement (strongly). Heating rates of the order of 10^3 °C per second are obtainable by this means.

It should be emphasized that the above described laser heating - even the non-hybrid direct case of CaSO_4 - is not an instance of optically stimulated luminescence; it is a means of heating the crystal by absorption arising from the host crystal lattice itself. Stimulated luminescence, on the other hand, requires absorption of light directly by the trapped-electron or trapped-hole centers, which would be expected to take place at considerably shorter infrared wavelengths. However, with the advent of tunable lasers covering a wide range of infrared, the possibility now exists of developing phosphors and light sources leading to practical stimulated luminescence dosimetry systems.

8.5 SUMMARY

High energy radiations can cause changes in the optical properties of solids as a consequence of the ionization and atomic displacement they produce. Some of these changes are sufficiently persistent to permit their employment as measures of the integrating radiation exposure of the solid. The most common of these radiation-induced effects is a change of optical absorption; if the absorption change lies in the visible spectrum and if the radiation exposure is

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of sufficient magnitude, color changes can be seen visually. At lower exposures or in other spectral ranges the absorption change is observable only instrumentally. Other optical effects which may accompany the absorption changes are new luminescence properties: "radiophotoluminescence," the ability of the solid to emit fluorescent light when illuminated with light lying in the newly created absorption band; or "thermoluminescence," the ability of the solid to emit light when it is warmed well below the temperature of incandescence. The physical processes of creating the new optical properties generally involve structural imperfections in crystals, principally vacant lattice sites, atoms in interstitial lattice positions, and chemical impurities. These imperfections react with the electronic charges freed by the radiation and form new non-equilibrium localized atomic configurations ("centers") which give the solid its new optical properties. The movement of free electronic charge through the crystal can be understood in terms of the energy-band model of solids, as can the energetics and kinetics of capture ("trapping") and liberation ("untrapping") of these charges by the imperfections. The specific properties of the radiation-induced centers determine the mode of their utilization in dosimetry. The two main practical cases are: 1) centers which emit luminescence themselves when excited by light they absorb (the basis for radiophotoluminescence dosimetry) and 2) centers that liberate their trapped charge on moderate heating, giving up the stored energy to other entities in the solid which, then emit their own luminescence (the basis for thermoluminescence dosimetry).

Prototype and practical systems of both kinds are described. The performance of the most important practical systems is surveyed, including the sensitivity (dose range), energy-dependence, storage stability, and other important properties of the dosimeter solids, as well as the design of selected dosimeters and reading instruments.

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CHAPTER 9

FILM DOSIMETRY

L. RADZIEMSKI

9.1 PRINCIPLES OF OPERATION

Film dosimetry, the measurement of dose by analysis of the blackening of photographic film, is the oldest form of personnel dosimetry. Silver halide photographic emulsions themselves have been used in nuclear physics since Henri Becquerel discovered in 1896 that the radioactive decay of uranium blackened photographic plates. The early emulsions had silver halide grains so large that particle tracks could not be resolved. In the early 1930's special emulsions with small grains were developed which allowed the observation of individual nuclear tracks. Despite the age of the technique it is still viable and useful, principally because of its simplicity and the films permanence. The ability to have a permanent record of a person's dose history makes it attractive. Interestingly enough, it may be economics and not science which will make it a less attractive method in the future. The high price of silver is presently leading film users to seek alternative dosimetry methods which will keep costs down. The most extensive use of nuclear emulsions today is for film badges that monitor gamma and x-ray dose. There is a large body of literature describing the use of photographic film in nuclear applications and dosimetry. One of the first quantitative papers was "The Sensitivity of Photographic Plates to x-rays" by Allen and Lafy in 1917. Good discussions of the subject are found in books by Becker (1966, 1973), a review by Dudley (1966), a book by Herz (1969), and an article by McLaughlin (1970). An important source of information on nuclear tracks in solids is Fleischer et al., (1975). A more recent discussion of nuclear track emulsions is given by Erskine (1979), and brief summaries are found in books by Knoll (1979) and by Eichholz and Posten (1979). The latter books also discuss the new technique of polycarbonate film dosimetry.

We will discuss properties of the various types of films, silver halide photographic, nuclear track, and polycarbonate, their recommended method of use and calibration, and summarize their advantages and disadvantages. The newest concepts in neutron film dosimetry and spark counting will also be considered.

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9.1.1 Exposure Theory

The photographic process uses the fact that light and radiation cause irreversible chemical changes in certain sensitive molecules. The detecting element in the photographic process is a small silver bromide or silver chloride crystal or grain. These grains, suspended in a gel, are coated on a clear cellulose-acetate or polyester substrate, occasionally on a glass plate. The sizes of the grains helps determine the resolution of the film. Coarse grained emulsions ($\sim 1\mu\text{m}$) are sufficient for conventional dosimetry. Finer grained films ($0.25\mu\text{m}$) are needed to see short and narrow nuclear tracks. When struck by radiation, either photons or charged particles, the silver halide grains are converted, after suitable chemical processing, into visible lumps of silver. In the development process only the crystals struck by primary or secondary radiation are changed to silver specks. Development enhances the contrast between the exposed and unexposed areas of the film. With careful exposure and processing the amount of film blackening (or concentration of tracks) can be related to absorbed energy.

The theory of the photographic process is based on the complex theory of the latent image. This image is formed by silver atoms gathered in sensitive lattice sites in an AgBr crystal. These neutral atoms are thought to result from the recombination of silver ions with migrating electrons liberated by the incident radiation. The chemical processing of the photographic material amplifies the latent image by nucleating the silver atoms and dissolving away the remaining, unirradiated AgBr grains. Care must be taken to avoid over development which is the reduction of all of the Ag^+ ions to silver atoms, even those that have not been exposed to radiation. The details of film dosimetry are well known, the definitive work on the subject being that of Mees (1954), a volume edited more recently by James (1977). The development is terminated by a stop bath with changes the pH of the solution. There undeveloped grains are washed away by a hyposulfite "fixer." Finally, the film is washed in fresh water and dried in the air. Storage which avoids film abrasion is desirable. Useful information on films and their processing characteristics can be found in commercial handbooks (Kodak, 1967 and subsequent years).

9.1.2 Properties of Photographic Films

Before discussing the relation between film blackening and radiation exposure, we must introduce a number of concepts common to the field of photography and films (see Altman (1977)).

Exposure (photographic) is the product of intensity and time, and represents the power per unit area of the emulsion. Any other unit proportional to power may also be used, R, for example when the energy is in the form of nuclear radiation. The photographic concept of exposure should not be confused with the radiological use of the

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word. Significant differences between the effect of exposure of film to ionizing radiation and to light are discussed by Broadhead (1977).

The quantitative measure of film blackening is called optical density (specifically, diffuse optical density). If light of intensity I_0 is incident on a film, and intensity I is transmitted, the optical density is

$$OD = \log_{10} \frac{I_0}{I} \quad (9.1)$$

Hence an optical density of 1 corresponds to 10% transmission, $OD = 0.3$ to 50% transmission. An emulsion which is developed even without being exposed to light will have a background fog of 0.01 to 0.02 optical density units. These must be subtracted from optical densities caused by irradiation. Diffuse optical densities can easily be measured in the range from 0.01 to 4.0 by most quality densitometers.

The most important quantitative property of an emulsion is the characteristic curve, also called the optical density log-exposure, Hurter and Driffield, or H and D curve. It relates the quantity of exposure on a logarithmic scale to the optical density of the developed image. Figure 9.1 shows a typical curve. The lower left is the toe or threshold region, the middle part is the linear region, and the upper right the saturation region. Most accurate measurements are made while working on the linear portion of the semilog curve.

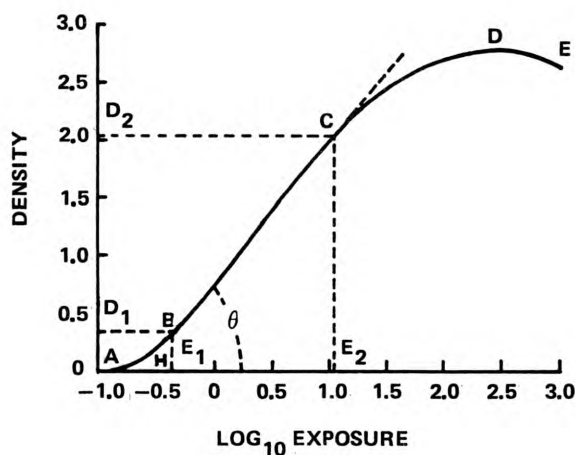


FIGURE 9.1 THE CHARACTERISTIC CURVE FOR A TYPICAL PHOTOGRAPHIC EMULSION (SAWYER, 1963).

The emulsion sensitivity is the measure of the response of emulsion to radiation. It is defined as the reciprocal of the exposure needed to reach $OD = 0.3$. Hence, the less exposure needed to reach that optical density, the greater the sensitivity.

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Occasionally the sensitivity for a dosimetric film is defined in terms of the absorbed dose needed for a given optical density. (See Chapter 1 for definition of absorbed dose.)

The slope of the density-exposure curve gives the emulsion contrast. The higher the slope, the smaller the range of exposure needs to go from low to high density. In Figure 9.1, the contrast is $\tan \theta$. Related to this is the film latitude, the ratio of maximum to minimum exposure to which the film is sensitive. The latitude is D_2/D_1 in Figure 9.1.

In principle, the response of an emulsion depends only on the exposure, the intensity-time product. In fact, a low intensity-long exposure does not normally result in the same optical density as a high intensity, short exposure. This is known as reciprocity failure. Reciprocity generally holds for ionizing radiation, but not for visible photon irradiation at very low or very high exposure levels. Images lose their optical density as a function of time after processing. This is known as fading and must be taken into account when looking at films when there is a long period between irradiation and processing. Other characteristic curves can be found in the Kodak handbook, Mees (1954), and James (1977).

9.1.3 Nuclear Track Films

Nuclear track emulsions are often used for neutron dosimetry. These films, such as Kodak NTA, differ from conventional emulsions in the emulsion thickness and silver halide loading. The thickness is increased to as much as 500 micrometers to allow the recording of entire particle tracks. In order to increase the film resolution the concentration of silver halide in the emulsion is increased to as much as 80%. More elaborate procedures are needed to insure uniform development of these thicker emulsions. A complete nuclear track can give information on the particle range or energy from its length and the type of radiation from its thickness.

Fast neutrons generate recoil protons through the $^{14}\text{N}(n,p)^{14}\text{C}$ reaction and similar n,p reactions, and it is these recoil protons whose tracks are captured in the emulsion. Such reactions are similar to those resulting in energy deposition in tissue. Lehman (1961) and Piesch (1963) characterized emulsion sensitivity versus neutron flux, and found that generally fewer protons were produced per unit neutron flux as the neutron energy increased. The sensitivity of track emulsions to thermal neutrons can be increased by loading the emulsion with special target nuclei such as boron or uranium. The former generates particle tracks through $^{10}\text{B}(n,\alpha)^7\text{Li}$ and the latter from $^{235}\text{U}(n,f)$ reactions.

9.1.4 Track-Etch Detectors

Conventional silver halide nuclear track emulsions used for neutron dosimetry have a number of disadvantages. The more important ones are the sensitivity to other types of radiations, fading, and limited shelf life. These have driven research in the direction of finding better neutron dosimeter systems. One class of these are track-etch detectors. The principles are simple. A material which has nuclei that react with neutrons will suffer track damage when placed in a neutron field, but will be relatively insensitive to photons and betas. In themselves, the tracks are small and in some materials subject to fading and "annealing" due to temperature effects. Etching makes the tracks larger and easier to count. Although many inorganic materials have been used in this way (Eichholz and Poston, 1979) we will only describe in detail polycarbonate foils and chemical etching. It should be noted that each track etch system shows advantages for certain applications. The basic reference in this field is Fleischer et al., (1975).

Track damage in polycarbonates is thought to occur due to breaking of long polymer chains in the material. The number of tracks is proportional to the radiation flux and thus to the absorbed dose, as in the case of nuclear track emulsions. But by contrast with track emulsions, where the tracks are strings or lines of grains, here the tracks are physical erosions of the surface or body of the material itself. As emulsion tracks can fade with time, track damage can also fade, or be "annealed" at elevated temperatures. Fading cannot be inhibited for emulsions except by controlling the storage environment. The great advantage of tracks recorded by material damage is that they can be enlarged by etching. For example, pits at the site of particle-track entrance in cellulose acetate pits are etched for 30 minutes in a 28% solution of KOH at 60°C, and cellulose nitrate by treatment in the same solution at 28°C or 6.25 normal NaOH at 70°C (2 min.). The optimum etching time is a function of track density and type of irradiation. At high track density, extended etching causes adjacent holes to overlap. With mixed radiation fields tracks due to the different components can be uncovered with different etching times.

A recent development in etching technique is the electrochemical etch, first discussed by Tommasino and Armellini (1970). In this method, a spark through the etchant and damage track results in preferential local etching along the track at a rate much greater than the bulk etching rate, leading to very large tracks, round in cross section, and easy to measure. This technique seems to be an intermediate step to a promising new development in neutron dosimetry, namely the extension on the breakdown detector to a realtime thin film capacitor counter (Tommasino et al., 1977).

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9.2 NORMAL OPERATIONAL USE

There are many elements which go into making up a system for dosimetry. These range from the choice of film for the particular application, to its packaging in a film badge, to its positioning on the body, processing, and last but not least, calibration. It is impossible to treat all the types of systems or situations encountered in the field. We will describe a few. Treatises by Becker (1966), Ehrlich and Fitch (1951) and Dudley (1966) provide details and additional references.

9.2.1 Choice of Film and Packaging

In either personnel monitoring or special applications the choice of film is crucial. Each film has a range of exposure to which it can be made to respond by variations in processing. Examples of these are shown in Figure 9.2 (Becker, 1966) where characteristic curves for many silver halide emulsions are summarized. The total range of exposure cannot be covered by any single emulsion but a reasonable range can be covered by two. Note that the films cited cover a dose range from 10^{-2} to 10^4 R when the films have readable

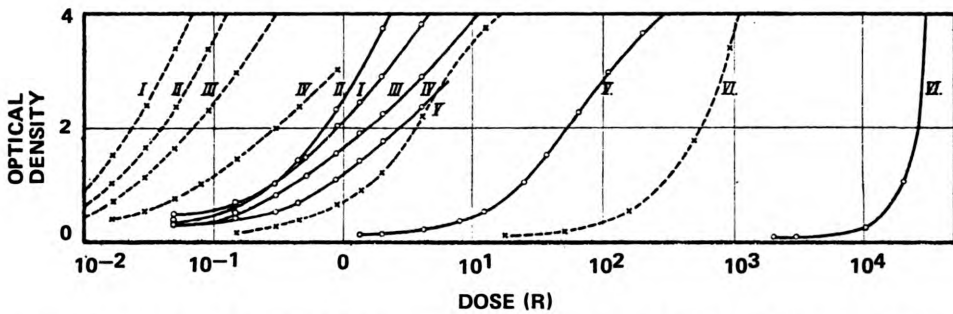


FIGURE 9.2 CHARACTERISTIC CURVES OF VARIOUS EMULSIONS WITHIN THE RANGE OF MAXIMUM (—) AND MINIMUM (---) SENSITIVITY. (I) AGFA SINO; (II) ADOX HIGH-SPEED DOSIMETER FILM; (III) KODAK RADIATION MONITORING FILM; (IV) GEVAERT DENTAL FILM; (V) ADOX LOW-SPEED DOSIMETER FILM; (VI) AGFA MIKRAT PLATE (BECKER, 1966).

optical densities between 1 and 4. For extended range coverage, combinations of emulsions are used according to one of the following methods. First, it is possible to coat different areas of the same film-base with a different sensitivity. Finally, two or more film packs may be placed in the dosimeter. Table 9.1 (Kiefer and Maushart, 1972) summarizes some of the configurations that have been used. Figure 9.3 (Becker, 1966) illustrates the range of dose that can be covered by these and other film sensitization techniques.

Some of the considerations in film wrapping and packaging are protective and some are political or economical. In general the film must be protected against accidental exposure to light and excessive humidity which alters the development characteristics. Uniformity

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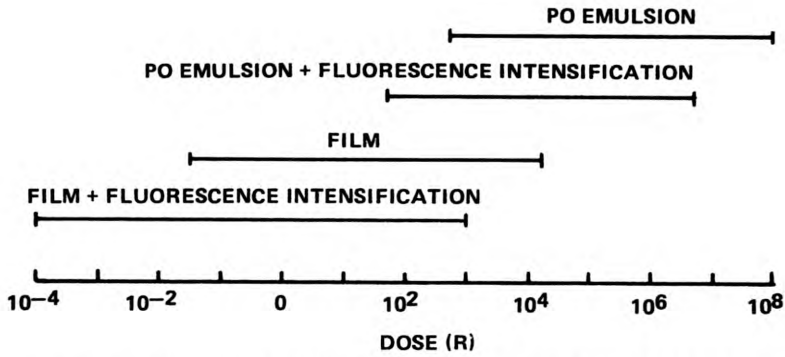


FIGURE 9.3 SENSITIVITY RANGES OF VARIOUS FILM DOSIMETRY SYSTEMS WITH REGARD TO GAMMA RADIATION (BECKER, 1966).

of wrapping is desirable if the material attenuates the radiation of interest. It is not easy to reconcile different requirements on the badge itself. Inexpensive and very stable badges are needed for civil defense dosimetry. Reactor operator badges should be able to measure neutron doses over wide ranges of energy. Medical installations use badges that are sensitive to very low absorbed doses and dose rates. One type of badge arrangement which has been used at Oak Ridge National Laboratory is shown in Figure 9.4 (Becker, 1966). A number of different types of detectors have been incorporated into this badge.

TABLE 9.1 EXAMPLES OF VARIOUS METHODS FOR OBTAINING ENERGY-INDEPENDANT DOSE RECORDING WITH FILM DOSIMETERS (KIEFER AND MAUSHART, 1972).

	Filter Combination		Film Combination van Stekelenburg (1958)	Fluorescence Compensation		Filter Analysis Dresel (1956)
		Ehrlich and Fitch (1962)		Hoerlin (1953)	Becker (1957)	
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Combination	Kodak PM1 1.2 mm Sn	Du Pont 510 8.25 mm Bakelite 1.07 mm Sn 0.3 mm Pb	Ilford PM 3 2 mm Sn Ilford PM 1 2 mm Sn	Special film p-terphenyl pressed emulsions	Agfa-AGEPE p-terphenyl emulsion 1 mm Cu	Adox film, filter-free region 0.05 mm Cu 0.5 mm Cu 1.2 mm Cu
Energy Range	0.100-1.2 MeV ± 20%	0.115-10 MeV ± 21%	0.030-1.2 MeV ± 20%	0.040-1.2 MeV ± 15%	0.110-1.2 MeV	0.030-1.2 MeV energy-dependent
Advantages	Simple arrangement	Electron equilibrium up to 10 MeV	Energy- independent	Nondirectional, dose sensitivity up to mR-range	Intensity-dependent fading	Energy determination (in homogeneous radiation, etc.)
Disadvantages	Direction dependence	Direction dependence	Direction dependence Evaluation of two films			Direction dependence, complicated handling because of graphical evaluation and 3 dose fractions

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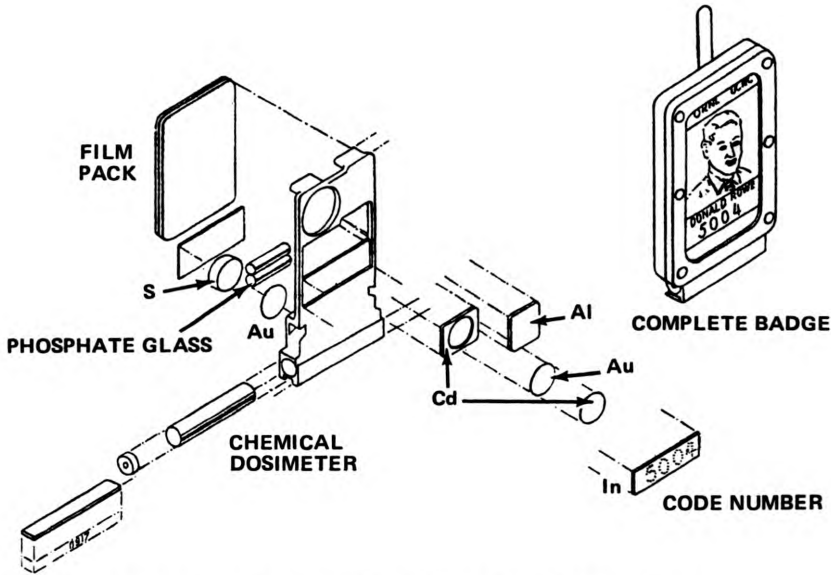


FIGURE 9.4 OAK RIDGE DOSIMETER (MORGAN, 1961).

9.2.2 Processing Technique for Silver Halide Emulsions

The task of processing silver halide dosimetric film and assessing doses is normally carried out by a laboratory specializing in such services. Such an organization has the advantage of standardized techniques for calibration and processing, and central recording facilities. As an example, the central processing laboratory for the Air Force is located at Brooks AFB, and is the USAF Occupational and Environmental Health Laboratory. Since procedures vary with organizations we will only outline some of the more crucial steps in the processing. These are schematically shown in Figure 9.5 (Becker, 1966).

Prior to use, dosimetric silver halide film should be stored in a dry atmosphere at cool temperatures and at relative humidities of less than 50%. Careful sealing will reduce latent image fading due to moisture (Ehrlich, 1962). After development and collection, badge and film should be checked for contamination before processing. Darkroom illumination with low level red light is permitted during development. Constant development conditions should be a goal as departures from these influence the film characteristic curve. Calibration films should be developed under the same conditions, indeed at the same time, as the exposure dosimeter film. The effect of departures from these conditions is discussed in detail by Becker (1966).

When film response to radiation is being measured in terms of increase of optical density a good densitometer is required. At low optical density the measurement accuracy should be 0 ± 0.02 density units. For a variety of films, this corresponds to the range ± 6 to $\pm$

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40 mR. At the high end densities of 4 to 6 should be readable. Some densitometers are automated and record data on magnetic tape (Steinhaus et al., 1971; Schneider and Klystra, 1970).

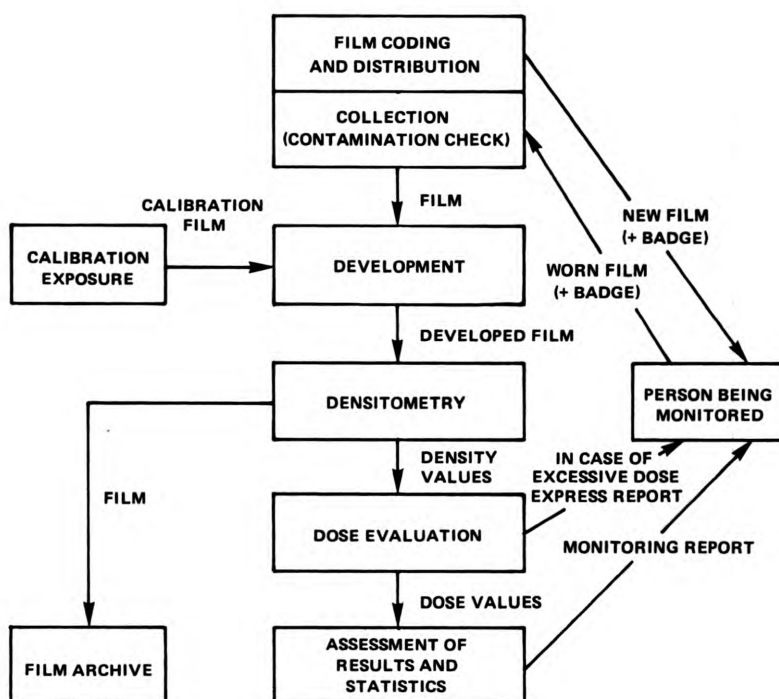


FIGURE 9.5 SCHEMATICAL ORGANIZATION DIAGRAM OF A FILM BADGE SERVICE.

The measurement of nuclear track emulsions is normally made with a microscope, which is a much slower and tedious task. Connection of a scanning densitometer to a digital computer and the use of a pattern recognition code has reduced much of this labor. Erskine (1979) discusses a number of scanning machines developed in the late 60's and early 70's for this purpose.

9.2.3 Track-Etch Film Processing

The processing of track-etch detectors, of which polycarbonate films are one example, is relatively simple. The exposed film is placed in the etchant of specific chemical composition for a known period of time at a known temperature. By analogy with film processing, the concentration of the etchant, temperature, and time are the parameters which must be well regulated to insure reliability and reproducibility. After etching, the enlarged tracks can be read on microscopes by trained operators, or by computerized scanners.

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9.3 CALIBRATION

Photographic films do not respond identically to all types of radiation. Hence the same film blackening can imply different doses depending on the type of radiation interacting with the film. For electrons, the chemical composition and thickness of the emulsion are important because they differ from the composition of tissue. The Compton process is most important for gelatin and tissue but in the emulsion the photoelectric effect dominates at low energies and pair production at high energies. The neutron dosimetry problem is equally complex because neutrons are always accompanied by a background of gamma rays. Because of these problems we will discuss the calibration of silver halide film response to the different types of radiation separately from calibration of track-etch detectors.

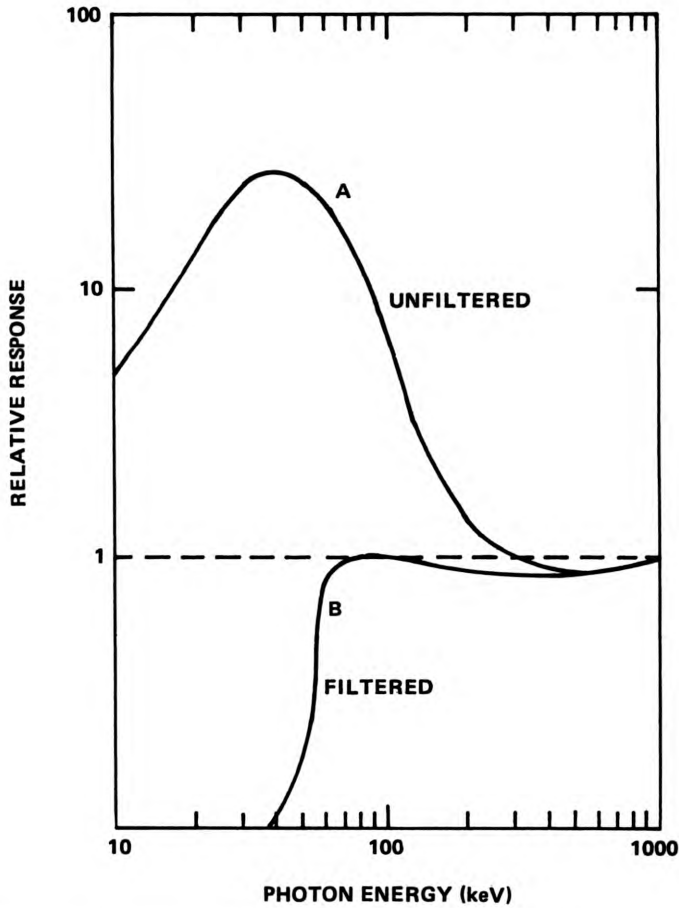


FIGURE 9.6 RESPONSE OF A TYPICAL FILM BADGE DOSIMETER WITH AND WITHOUT A COMPENSATION FILTER (AFTER HEARD, 1965).

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9.3.1 Film Calibration for X and Gamma Rays

Photons produce energetic secondary electrons which in turn produce a latent image in the emulsion. This happens by three processes: Compton, photoelectric, and pair production. For silver bromide and silver chloride the photoelectric process dominates at low-energy and pair production at high energy. For gelatin and tissue the Compton process is the most important. In the photoelectric effect, large amounts of energy are deposited by low energy-photons, which leads to a drastic variation in film sensitivity with photon energy. This is illustrated by the upper curve in Figure 9.6 (Knoll, 1979). The effect is entirely analogous to the wavelength dependence of emulsion sensitivity for visible and infrared radiation.

Among the methods to correct for, or evaluate this effect are:

1. The use of supplemental absorbers.
2. The use of auxiliary scintillators.
3. Correction for the photon energy spectrum.
4. Manipulation of emulsion characteristics.

Supplemental absorbers attenuate the low energy photons to reduce their effect, as shown by the lower curve in Figure 9.6. Scintillators amplify the effects of high energy photons. Emulsion modifications have been investigated by Hoerlin (1953). Two methods have been used to determine photon energies, the ratio of penetration through different filter thicknesses and the relative photographic action of secondary electrons ejected from different metal foils. In general, the addition of filters to film dosimeters increases their dependence on the angle of incidence, not otherwise a great factor.

The calibration of film response and badge effects is treated in laboratory manuals by Oregon State University (1973) and Gollnick (1978). Calibration may be performed with the following apparatus:

1. X-ray unit or ^{137}Cs , ^{60}Co , or ^{226}Ra sources with beta-ray filters.
2. Nine personnel monitoring film packets.
3. Densitometer.
4. Film hanger.
5. Two 0-200 mR pocket chambers and charge reader.
6. One film processing station.
7. One lead sheet.

Procedure:

1. Number the films in the upper right hand corner from one to nine.

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2. Make the following exposures with the film packets placed on the center of the lead sheet with a pocket chamber on either side to measure the exposure. Record the average of the pocket chamber readings.

3. If x-ray machine is used: 60 kVp, 1.0 mA, 3mm Al added filtration.

4. Film exposures: (film number) $\times$ 15 mR approximate, films 1 to 8. Film nine, no exposure, background control.

5. Develop the nine films: develop for five minutes at 68°F, rinse for 30 seconds, fix for 10 minutes, wash for 30 minutes, dry.

6. Read optical densities with densitometer and record.

7. Calculate net densities by subtracting the control film density from each of the other film densities.

8. Plot net density vs. exposure on graph paper. This gives the calibration curve for the film used under the development conditions employed, and for the geometry employed in the calibration.

9.3.2 Calibration for Electrons

The sensitivity of emulsions to energy deposition from electrons is also energy dependent. For a variety of emulsion combinations, the sensitivity is shown in Figure 9.7 (Dudley, 1966). The toe region results from the inability of the electrons to penetrate the emulsion. For electrons of medium energy, the sensitivity peaks because they traverse a longer distance in the emulsion. High energy electrons can pass right through without depositing much energy, so at this point the sensitivity flattens out. With these curves it is possible to calibrate responses to monoenergetic electrons. For a beta-ray spectrum, the spectrum composition must be determined by other means, and then the monoenergetic response weighted accordingly.

Film sensitivity depends on the electron angle of incidence, and so on the source position. For example, high energy electrons will deposit more energy in the emulsion if they traverse it at nearly grazing incidence rather than at normal incidence. Studies by Dudley (1954) showed a sensitivity change by a factor of 2.6 between 17 and 53 degree angle of incidence. A detector far away from a source sees electrons incident at a nearly constant angle, while one immersed in a radiation field sees all angles of incidence. The surface and volume effects are different for the two cases, and their extension to tissue absorption also changes. Details of such considerations are given in Dudley (1966).

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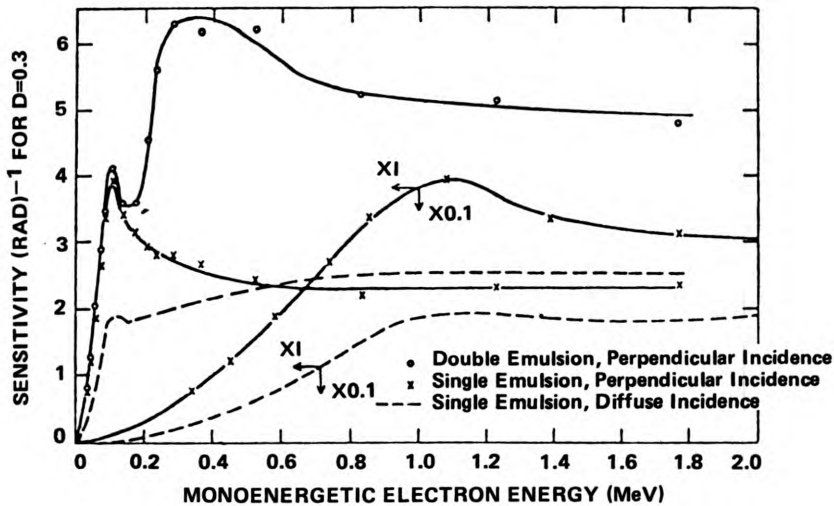


FIGURE 9.7 SENSITIVITY OF EASTMAN NO. SCREEN FILM FOR MONOENERGETIC ELECTRONS OF VARIOUS ENERGIES (DUDLEY, 1954).

Film calibration for incident electrons can be done using the same procedure as in Section 9.3.1, except that the source should be a standard electron source such as ^{90}Sr or ^{36}Cl , a known distance from the film. In the absence of a standard source, beta-detectors should be placed to either side of the film.

9.3.3 Calibration for Neutrons

The same $^{14}\text{N}(n,p)^{14}\text{C}$ reaction that results in an important absorbed dose in tissue is the one which produces the recoil protons which result in much of the film blackening due to neutrons. The addition of lithium or boron to the emulsion increases the energy transfer because of their large cross section for thermal neutrons. Energy is also deposited by the radioactivation of atoms in the emulsion. The atoms subsequently decay by beta and gamma-ray emission. The silver isotopes produced, ^{108}Ag and ^{110}Ag , contribute over 90% of the energy absorbed by the film due to this process.

The fundamental problem in neutron dosimetry is that there is very little energy deposited in the film, so that its direct response is almost negligible. Metal foils are often placed next to the emulsion to intensify the response by induced secondary radiations. Most commonly used are thin foils of cadmium.

Analysis of neutron dose by measuring the optical density on photographic film is not very practical. The accompanying gamma rays contribute a background fog, and it is hard to pick out the neutron effects. The sensitizer foils place another uncertainty in the response calibration procedure. It is better to measure the neutron

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dose by particle track analysis.

Calibration of film blackening by neutron irradiation is done by comparing with the effect of a standard neutron source at a known distance. Typical standard sources are ^{252}Cf and $^{239}\text{Pu/Be}$.

9.3.4 Calibration of Nuclear Track Emulsions

The number of tracks per cm is related to the dose by using a neutron source of known strength at known distance. Table 9.2 indicates the result of one calibration for Kodak NTA film (Becker, 1966). It shows the number of tracks per neutron and per cm^2 rad for three categories of neutron energy. One must be aware of the limitations of this method. The true length of tracks is measured only for tracks parallel to the film surface; otherwise they are foreshortened. This is important if the energy of the incident particle is a consideration. Fading, which depends strongly on the water content of the emulsion, hence on the humidity, may reduce the grain density until the individual tracks are unrecognizable. Neutrons with high energy may traverse the emulsion without depositing any energy.

TABLE 9.2 CALIBRATION OF KODAK NTA TYPE FILM.

NEUTRON ENERGY	NUMBER OF TRACKS PER NEUTRON	NUMBER OF TRACKS PER cm^2 RAD
FISSION SPECTRUM	3.8×10^{-4}	13.6×10^{-4}
Po-Be (APPROX. 4 MeV)	6.2×10^{-4}	15.5×10^{-4}
14 MeV	6.5×10^{-4}	10.5×10^{-4}

There are other practical problems. The angular dependence of the incident radiation affects the sensitivity, as in the case of electrons. For neutrons between 0.2 and 0.4 Mev the track length drops to about $3 \mu\text{m}$, comprising only a few grains in the finest emulsions. On the other hand, when there are a reasonable number of tracks to be counted, their counting is tedious and time-consuming unless automated methods are used as mentioned above.

9.3.5 Calibration of Track-etch Systems

Track-etch systems are calibrated like nuclear track emulsions, by using a standard neutron source at a known distance. They are less prone to error in counting, however, because of the size of the resultant holes.

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9.3.6 New Concepts in Calibration and Counting

When chemical etching of films, which results in relatively large holes, is combined with spark counting, a new and fast method of analysing track records is obtained. If the perforated film is placed in an electric field, sparks go through the enlarged holes. The number of sparks produced corresponds to the number of holes in the film and are counted by scalar. Multiple sparking and short circuits are avoided because each spark evaporates a large hole in the plate electrode. A schematic diagram of the spark chamber is shown in Figure 9.8 (Griffith et al., 1979). The spark counting technique has its own disadvantages for fast neutron measurements. The etching and counting conditions must be carefully controlled to maintain reproducibility. If the track density gets over $\sim 3000/\text{cm}^2$ the holes start to overlap causing saturation error. The polymer film dielectric properties must be well known because the applied voltage and breakdown characteristics depend on them.

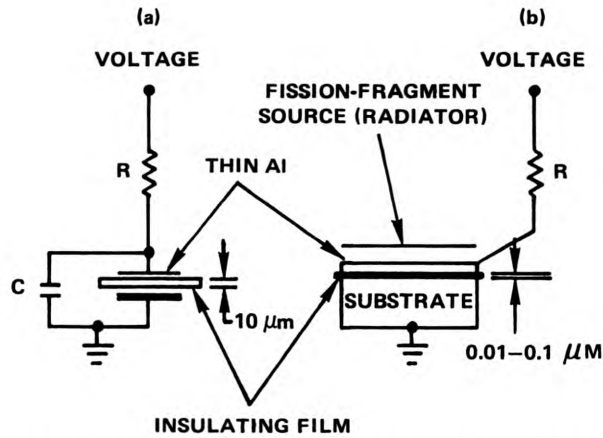


FIGURE 9.8 SCHEMATICS OF SPARK-REPLICA COUNTER AND BREAKDOWN DETECTOR (GRIFFITH ET AL., 1979).

The response of the system to high-energy neutrons is calibrated by using a standard source at a known distance. Accuracy of measurement has been estimated by Griffith, et al., (1979) to range from $\pm 28\%$ for low exposures in the 40 mrem range, to $\pm 7\%$ for high exposure of 1500 mrem and greater.

9.3.7 Sources of Error and Estimates of Accuracy

Correct assessments of film badge doses are extremely important. A repeated unusual reading can lead to complacency or serious misunderstanding if the limitations of the dosimetric method are not well understood. Principal categories of errors are listed

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below followed by a discussion of practical accuracy.

1. Insensitivity near the lower detection limit. The smaller the optical density, especially in the presence of background fog, the greater the inaccuracy. A factor-of-two error in gamma ray dose in the 10-100 mR range is quite likely.

2. Energy dependence. Even if well compensated for, correction for energy distribution of the radiation introduces additional errors, in the 10-20% range. Larger errors may be introduced by inaccurate assessment of the energy distribution itself.

3. Inaccurate estimation of beta-ray and neutron doses. These are the doses subject to greatest interference in the presence of a mixed radiation field. Low energy gamma-rays can interfere seriously with 0.4 eV to 0.4 Mev neutrons.

4. Emulsion calibration. Many errors and uncertainties contribute here; latent image degradation, calibration, uneven coatings, development.

5. Gases and vapors on the emulsion. All of the errors above and others listed in Becker (1966), Chapter VI, deal with the problem of estimating absorbed dose from optical density. In addition to that is the question of estimating the actual biological effect from the radiation dose. Passage of radiation through the body (neutrons) will generally cause an underestimate of the dose. Assymetrical arrangement of filters have been used to compensate for this effect. Others calculate a 10 to 40% underestimate of dose from this cause.

TABLE 9.3 CHARACTERISTICS OF SOME DOSIMETRIC FILMS

FILM	RADIATION	ENERGY RANGE	EXPOSURE RANGE	UNCERTAINTY
STANDARD	X, GAMMA	< 100 keV	10 mR - 60 R	>±30 mR or 20%
RADIOMETRIC	BETA	>100 keV	10 mR - 500 R	>±20 mR or 10%
FILM (KODAK)	BETA	>1.5 keV	40 mR - 1000 R	>±30 mR or 20%
NTA FILM (KODAK)	NEUTRONS			
NEUTRAK I (LANDAUER)	NEUTRONS	1-14 Mev	20 mR - 25 R	>±30 mR or 30%
NEUTRAK 144 (LANDAUER)	NEUTRONS	144 keV -	10 mR - 20 R	>±20 mR or 20%

Errors will sometimes lead to overestimation of dose and sometimes to underestimation. Under the best conditions, and with gamma- and x-ray radiation, we can normally assume that a dose will be

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determined to $\pm 20-50\%$. Gorson et al., (1965) discusses the reliability of measurement services. Taking the average of 12 companies, he found 71% of the values were correct within $\pm 50\%$, 81% within $-50/+200\%$. Table 9.3 summarizes the calibration results for a number of commercial films and systems.

9.4 SUMMARY

We have considered three basic systems in this chapter: dosimetry based on the blackening of photographic silver-halide film, dosimetry based on nuclear track analysis, and new techniques exemplified by polycarbonate films. Their advantages and disadvantages will now be summarized.

The film dosimetric system based on measurement of blackening of a photographic emulsion is easy to use, requiring little attention from the experimenter while the data are being collected. The emulsion takes little space and no external support system. It has its own memory device, which is only slightly affected by fading. The film detector is continuously sensitive and does not require an external power source. It has an inherently large dynamic range, being able to measure from tens of mR to hundreds of thousands of R with special techniques. And the cost is modest compared to other costs in nuclear experiments.

The principle disadvantage of this system is the delay in obtaining the dose information. The detector is not a real-time apparatus. Also, there is no time discrimination for events as long as the emulsion is exposed to radiation. An external shutter would be the only way to record time information. Silver halide emulsions do not work well with particles of very short or very long range. Very thin or thick emulsions would be needed to handle such cases. The method is well suited to low-level, long-term personnel monitoring for exposure to x-rays, gamma rays, and electrons.

The nuclear track dosimetric method has all the advantages listed above for the densitometric method plus a few; however, it is limited to neutron dosimetry. The nuclear emulsion is superior to any electronic device for locating the path of a nuclear particle. It provides for direct visualization of events which in turn can lead to particle identification. The NTA emulsion has good response to high-energy neutrons, but is poor for low-energy neutrons. Accompanying low-energy neutrons and x rays tend to fog NTA film. The major problem with track recording is the work that must be done to analyze the tracks. The emulsion is usually searched with a microscope since the tracks are so small. Scanning machines are complicated and expensive. The principle application for this method is for monitoring of personnel in high-energy neutron environments.

The new track-etch techniques involving plastic films for neutron monitoring and spark counting can be considered to be in the

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research and development stage. They bypass some of the problems of the photographic method. Their energy threshold discriminates against low energy photons and electrons. A permanent non-fading record is created. The tedium of counting is reduced. But these seem to be intermediate steps to a promising new development, the extension of the breakdown detector to a realtime thin film capacitor counter (Tommasino et al., 1977). If such a development proves feasible, track dosimetry may evolve as a modern technique of radiation dosimetry.

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CHAPTER 10

NEUTRON ACTIVATION FOILS

R. F. FLEMING

10.1 INTRODUCTION

10.1.1 General

The measurement of a neutron field using the activation technique is accurate, simple, and has the additional advantage of being independent of any gamma-ray field. Most elements have one or more isotopes which become radioactive when they undergo nuclear reactions in a neutron field. Some of these reactions, especially the neutron capture or (n,γ) reaction, can be induced by low energy, or thermal, neutrons. Others, especially those emitting other nucleons such as (n,p) , (n,α) , $(n,2n)$ or (n,n') , require high energy, or fast, neutrons to induce the activity. Thus, in addition to measuring the total intensity of the neutron field, it is possible using the activation technique to characterize the energy spectrum of the neutrons.

Since each radioactive nuclide produced has its own unique decay half-life and unique gamma and/or beta-ray spectrum, the technique is quite specific, and in some cases, more than one useful activity can be measured from the irradiation of a single element. Since the activating element can exist in the form of a metal foil or wire, a compound, a liquid or even a gas, the measurement can be designed to determine the neutron field in precisely the volume of interest. Additionally, since the activating material can often be made quite small, the measurement process can have minimal perturbation of the field being measured.

Several general references exist which provide much more detailed discussion of the technique: Beckurts and Wirtz (1964), the IAEA Technical Report 107 (1970), and the manual Neutron Activation Foils (1973). Of especial importance is the Nuclear Standards section, Part 45, of the Annual Book of ASTM Standards (1982) which describes in detail the procedures for neutron fluence measurement in general, and many of the important specific applications. It is continually updated and is intended to represent the state-of-the-art at any time. Some the the relevant standards are listed in Table 10.1. Familiarity with these standards is necessary for any serious worker in this field.

NEUTRON ACTIVATION FOILS

TABLE 10.1 STANDARD PROCEDURES FOR NEUTRON
FLUENCE MEASUREMENTS.

- E 170 Definition of Terms Relating to Dosimetry
- E 181 Analysis of Radioisotopes
- E 261 Measuring Neutron Flux, Fluence, and Spectra by Radioactivation Techniques
- E 262 Measuring Thermal Neutron Flux by Radioactivation Techniques
- E 263 Measuring Fast—Neutron Flux by Radioactivation of Iron
- E 264 Measuring Fast—Neutron Flux by Radioactivation of Nickel
- E 265 Measuring Fast—Neutron Flux by Radioactivation of Sulfur
- E 266 Measuring Fast—Neutron Flux by Radioactivation of Aluminum
- E 343 Measuring Fast—Neutron Flux by Analysis of Molybdenum—99 Activity
from Uranium—238 Fission
- E 393 Measuring Fast—Neutron Flux by Analysis of Barium—140 from Uranium—238 Fission
- E 419 Guide for Selection of Neutron Activation Detector Materials
- E 481 Measuring Neutron—Flux Density by Radioactivation of Cobalt and Silver
- E 482 Recommended Practice for Neutron Dosimetry for Reactor Pressure Vessel Surveillance
- E 496 Measuring Neutron Flux Density and Average Energy from $^3\text{H}(\text{d},\text{n})^4\text{He}$ Neutron
Generators by Radioactivation Techniques
- E 522 Calibration of Germanium Detectors for Measurement of Gamma—Ray Emission Rates
of Radionuclides
- E 523 Measuring Fast—Neutron Flux Density by Radioactivation of Copper
- E 526 Measuring Fast—Neutron Flux Density by Radioactivation of Titanium
- E 704 Measuring Fast—Neutron Flux Density by Radioactivation of Uranium—238
- E 705 Measuring Fast—Neutron Flux Density by Radioactivation of Neptunium—237
- E 706 Master Matrix for Light—Water Reactor Pressure Vessel Surveillance Standards
- E 720 Guide for Selection of a Set of Neutron—Activation Foils for Determining Neutron
Spectra Used in Radiation—Hardness Testing of Electronics
- E 721 Determining Neutron Energy Spectra with Neutron—Activation Foils for Radiation—
Hardness Testing of Electronics
- E 722 Practice for Characterizing Neutron Energy Fluence Spectra in Terms of an Equivalent
Monoenergetic Neutron Fluence for Radiation—Hardness Testing of Electronics
- E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance

10.1.2 Radiation Measurement

The accurate characterization of a neutron field requires a careful measurement of the radiation emitted in the decay the radionuclide produced. The gamma rays emitted are of discrete energies characteristic of the particular nuclide and are detected using either a NaI scintillator or a germanium semiconductor detector with an associated multi-channel analyser (MCA) for acquisition of the gamma spectrum. Unless the gamma spectrum is very simple and known to be due to only one radionuclide, the germanium detector is to be preferred because of its much better energy resolution and stability. In either case it is the area (total counts) under the photopeak at the appropriate energy that provides the quantitative data. Only if one has verified in every case that no activity other than that to be measured is present in the sample, should one risk applying gross gamma counting without an MCA.

In cases where greater sensitivity is required and especially for radionuclides which emit no gammas, beta measurement may be

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necessary. Beta particles are emitted as a continuum down to zero energy with only the maximum energy a characteristic of the particular radionuclide, and, in addition, they are scattered and absorbed much more readily than gamma rays. Therefore quantification by beta counting is always more difficult than gamma counting. As with gammas, spectrometry is superior to gross beta counting.

Among the excellent texts on radiation measurement are Knoll (1979), Tsoulfanidis (1983), Eichholz and Poston (1979), and NCRP Report 58 (1978). The ASTM Standard for the calibration and use of gamma spectrometers is E 522.

10.1.3 Theory

The reaction rate per atom for a particular reaction is given as

$$r(t) = \int_0^{\infty} \sigma(E) \phi(E, t) dE \quad (10.1)$$

where $r(t)$ is the reaction rate per atom at time t , $\sigma(E)$ is the reaction cross section for neutrons of energy E , and $\phi(E, t)$ is the neutron flux spectrum at time t .

The cross section is a measure of the reaction probability and has units of area. The neutron flux is the quantity to be determined and has units of neutrons/cm²/sec. In order to simplify the discussion that follows we will treat the case of the time independent neutron field $\phi(E)$; the extension to the time-dependent case introduces mathematical complexity but the principles remain the same. The reaction rate is often written in terms of the spectrum-averaged cross section and the total neutron flux

$$r = \langle \sigma \rangle \bar{\Phi} \quad (10.2)$$

where

$$\bar{\Phi} = \int_0^{\infty} \phi(E) dE \quad \text{and} \quad \langle \sigma \rangle = \frac{\int_0^{\infty} \sigma(E) \phi(E) dE}{\int_0^{\infty} \phi(E) dE}$$

It is the reaction rate per atom multiplied by the number of atoms present in the sample and integrated over the time of irradiation that determines the total induced activity that is measured. Using previously tabulated cross sections and irradiation materials of

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known mass the neutron flux is determined.

If, on the other hand, the sample and the neutron field were already well known, the measurement would yield a value of the spectrum averaged cross section. If the neutron field and the cross section were well known (or were made to cancel when compared to a standard) the amount of activating material would be determined. This is the basis of neutron activation analysis and is described in detail in De Soete (1972), for example. Familiarity with these allied fields of cross section measurement and activation analysis can provide insight into the method of foil activation.

10.2 NEUTRON FLUX AND REACTION CROSS SECTIONS

Before beginning the discussion of the detailed treatment of neutron activation we need to establish some simple approximations for the neutron flux and cross sections in the energy regimes of interest. The term "neutron flux" is used here in its conventional meaning and is shorthand for the preferred terms "neutron flux density" or the "neutron fluence rate" (see ASTM Standard E 170).

10.2.1 Neutron Flux

The direction of neutron travel is often treated as one or the other extreme case - the isotropic distribution for which all directions are equally likely, and the beam case for which all neutrons are travelling in the same direction. For the beam case it is often more convenient to measure the beam intensity rather than the beam flux. The reaction rate in a beam is:

$$R = \Phi \langle \sigma \rangle N^* = I \langle \sigma \rangle \frac{N^*}{a} \quad (10.3)$$

where Φ is the total neutron flux (n/cm²/sec), I is the neutron intensity (n/sec), N^* is the number of activating nuclei in the beam, and a is the area of the beam (cm²). Note that if a foil is larger than the beam, a measurement of the foil mass and area determines N^*/a which allows I to be measured without knowing a or the profile of the neutron beam. We will now consider the energy regimes of importance.

Source Neutrons - Most neutron sources produce neutrons of high energy, usually in the MeV region. These may be essentially monoenergetic sources such as the 14 MeV neutrons produced by the $D(T,n)$ reaction. They may be a broad, but well-defined, spectra such as that which results from fission reactions. Or, the neutron distribution may be complex and unique to that particular source as is the case for isotopic (α,n) sources. The fission neutron

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spectrum is often represented as

$$\phi(e) = C\sqrt{E} e^{-\beta E} \quad (10.4)$$

where the parameter β , which characterizes the mean energy of the distribution, depends on the fissioning nuclide, e.g., ^{235}U , ^{239}Pu , or ^{252}Cf .

Intermediate Energy Neutrons - As the source neutrons collide with atoms in the surrounding materials they lose energy and establish a "slowing-down" distribution. This distribution is usually complex, especially if strong neutron absorbers are present, but after a few collisions the energy spectrum can often be approximated as

$$\phi(E) = \phi_0/E \quad (10.5)$$

where ϕ_0 is a constant with the unlikely units of flux per unit $\ln E$ interval.

Thermal Neutrons - The slowing down process continues until the neutrons are at a low enough energy (~ 1 eV) that the colliding atoms begin to contribute energy to the neutrons in some collisions. If an equilibrium is reached with the thermal motion of the atoms, the spectrum of neutrons is referred to as thermal. This can be written as

$$\phi(E) = \Phi \frac{E}{E_T^2} e^{-E/E_T} \quad (10.6)$$

where $E_T = kT$ = characteristic Boltzmann energy. Of special importance is the energy corresponding to neutrons with a velocity $v_0 = 2200$ m/sec. For this velocity $E_T = 0.0253$ eV and $T = 293.6^\circ\text{K}$.

Cadmium Neutrons - The element cadmium (actually its ^{113}Cd isotope) has the very useful property that its neutron transmission through about a 0.1 mm thickness is nearly zero for energies below 0.5 eV and nearly unity above this energy. Since this is approximately the dividing line between thermal and intermediate energy neutrons, cadmium acts as a thermal neutron shield. The neutrons stopped by cadmium are called sub-cadmium neutrons and those which pass through are called epi-cadmium. Therefore by covering an activation foil with cadmium the activation by thermal neutrons can be eliminated.

10.2.2 Neutron Cross Sections

There are some general statements that can be made about neutron cross sections in the three energy ranges of interest.

Fast Neutron Region - For energies above 1 MeV most neutron

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capture (n,γ) cross sections are quite small and the activation reactions of interest are the (n,N) or (n,f) reactions for which either a nucleon $(n,p,\alpha,\dots)$ is emitted or fission takes place. These reactions often have an effective threshold of a few MeV. Hence each activation reaction samples only those neutrons above a certain energy. These cross sections are usually slowly varying functions of energy.

Intermediate Neutron Region - In the energy region from a few eV to a few hundred keV the neutron cross sections are usually very complex functions of energy with sharp resonances at certain energies. These large cross sections at known resonance energies can sometimes be useful in measuring the slowing-down neutrons at a particular energy. Often, however, the integral of the resonance region cross section above the 0.5 eV cadmium cut-off energy (E_{Cd}) is measured or calculated for an assumed $1/E$ neutron spectrum. This so-called resonance integral is defined as

$$I = \int_{E_{Cd}}^{\infty} \frac{\sigma(E)dE}{E} \quad (10.7)$$

Thermal Neutron Region - In the thermal neutron region the activation cross sections are usually (n,γ) or (n,f) reactions. Below the cadmium cut-off energy of ~ 0.5 eV, these cross sections are usually slowly varying and, in fact, most can be well approximated by a $1/v$ dependence where v is the neutron velocity, that is

$$\sigma(E) = \sigma_0 \sqrt{\frac{E_0}{E}} = \sigma_0 \frac{v_0}{v} \quad (10.8)$$

where $v_0 = 2200$ m/sec and $E_0 = 0.0253$ eV. It is conventional to express thermal neutron cross sections as σ_0 , the value of the cross sections at $v = v_0$.

For some reactions, the $1/v$ cross section shape continues well above 0.5 eV before any significant resonance structure occurs. It is therefore useful to compute the ratio of the resonance integral to the 2200 m/sec cross section for the $1/v$ case. By inserting Equation 10.8 into Equation 10.7 we find

$$\frac{I}{\sigma_0} = 2 \sqrt{\frac{E_0}{E_{Cd}}} = 0.45 \quad (10.9)$$

The ratio I/σ_0 is a measure of the resonance-to-thermal activation and is useful for comparing foil materials. Extensive tabulations and plots of neutron cross sections are available, the most complete

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being the series by Mughabghab et al. (1981). The IAEA Technical Report 156 (1974) is particularly useful in that it also gives cross sections averaged over a fission neutron spectrum. A compilation of carefully evaluated cross sections for those reactions of most importance for neutron dosimetry is also available as Magurno (1975).

10.3 ACTIVATION, DECAY AND COUNTING

It is necessary at this point to sketch the derivation of the relationship between the number of counts measured for a particular gamma ray and the neutron flux and other parameters which correspond to the measurement. Consider the case for which the neutron flux is constant and the "burnup" due to neutron absorption of both target and product nuclei can be ignored. Both these assumptions are often valid and exact treatment is only somewhat more complicated. If N^* nuclei are irradiated in a flux Φ to produce N radioactive nuclei with decay constant λ , we have

$$\frac{dN}{dt} = N^* \langle \sigma \rangle \Phi - \lambda N \quad (10.10)$$

where

$$N^* = \frac{\alpha m N_A}{A}$$

The number of activated nuclei present at the end of an irradiation of length τ is

$$N(\tau) = N^* \frac{\langle \sigma \rangle}{\lambda} \Phi (1 - e^{-\lambda \tau}) \quad (10.11)$$

Taking the end of irradiation as the $t = 0$ point, we obtain

$$N(t) = N(\tau) e^{-\lambda t} \quad 0 < t \quad (10.12)$$

Suppose now that G gamma rays of energy E are emitted per decay and that the detector efficiency is ϵ counts per gamma. If we acquire counts for a time beginning at time t after the end of irradiation, then the total number of counts is

$$C = \epsilon G \int_{t_1}^{t_1 + \Delta} \lambda N(t) e^{-\lambda t} dt = \epsilon G N(\tau) e^{-\lambda t_1} (1 - e^{-\lambda \Delta}) \quad (10.13)$$

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$$C = \epsilon G N^* \frac{\langle \sigma \rangle}{\lambda} \Phi e^{-\lambda t_1} (1 - e^{-\lambda \tau}) (1 - e^{-\lambda \Delta})$$

We now define the experimental quantity C_0 which is equal to known constants and the neutron flux Φ to be determined.

$$C_0 \equiv \frac{\lambda C e^{\lambda t_1}}{(1 - e^{-\lambda \Delta})(1 - e^{-\lambda \tau})} = \epsilon G N^* \langle \sigma \rangle \Phi \quad (10.14)$$

where $\lambda = \ln(2)/T_{1/2}$ (decay constant), G = number of gamma rays of energy E_γ per decay, a = isotopic abundance, N_A = Avogadro's number, A = atomic weight, Φ = neutron flux during irradiation, $\langle \sigma \rangle$ = spectrum averaged activation cross section, ϵ = detector efficiency for gamma rays of energy E_γ , C = net counts in the gamma-ray peak, m = mass of the foil, τ = irradiation time, t_1 = time from end of irradiation to start of count, Δ = live time of counting, and N^* = number of atoms of the activating isotope.

The important nuclear parameters can be found in Lederer et al. (1978), Kocher (1981), and Zijp and Baard (1981). A very useful compilation for estimating induced activity is Erdtmann (1976).

10.4 MEASUREMENT OF NEUTRON FLUX

It is clear from Equation 10.14 that if a foil of known composition (N^*) is irradiated in a field for which $\langle \sigma \rangle$ is known and then counted on a calibrated detector (ϵ) to yield a value of C_0 , then the flux Φ can be determined. This is straightforward except for the fact that $\langle \sigma \rangle$ be known. From Equation 10.2 this means that to determine Φ the neutron energy spectrum must first be known. The fact that the measurement of Φ can be no better than the value of $\langle \sigma \rangle$, and hence of $\phi(E)$, must not be forgotten. However, if foils of several different materials are irradiated, estimates of the shape $\phi(E)$ can also be obtained. We will now treat some simple examples. For more detailed procedures, see Table 10.1.

10.4.1 Total Neutron Flux

Writing Equation 10.14 in terms of the neutron flux gives

$$\Phi = \frac{C_0}{\epsilon G N^* \langle \sigma \rangle} \quad (10.15)$$

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The result is the "effective" neutron flux for the spectrum $\phi(E)$ used to compute $\langle\sigma\rangle$. This result is not particularly satisfying unless $\phi(E)$ is known with some confidence.

10.4.2 Fast Neutron Flux

A particularly simple case is that of measuring the uncollided neutron flux from a source for which $\phi(E)$ is well known. This could be the $D(T,n)\alpha$ reaction which yields neutrons with a narrow spread around 14 MeV or a fission reaction such as $^{235}\text{U}(n,f)$ or ^{252}Cf which yields a broad but well known spectrum with a mean energy of about 2 MeV. By using a threshold reaction which is not sensitive to the neutrons that have undergone collisions and lost energy, the value of $\langle\sigma\rangle$ can be reliably calculated and Φ obtained.

10.4.3 Resonance Energy Neutron Flux

Certain elements such as In, Au, and Mn have an epithermal cross section dominated by a single sharp resonance at a particular energy. Because of the very large cross section at the resonance a thin layer of the material is sufficient to stop nearly all of the neutrons at the resonance energy. Therefore if three foils of the same material are stacked together as a "sandwich" the inner foil will not be activated by resonance neutrons, since they are completely absorbed in the outer two foils. If a foil is activated at the center of the sandwich and another foil is activated without covers, the difference in their activities will be a measure of the resonance flux. Suppose the two foils are chosen to have the same weight and that C_o^b is obtained for the foil irradiated bare and C_o^s is obtained for the foil irradiated in the sandwich. Then

$$\Phi(E_r) = \frac{C_o^b - C_o^s}{\epsilon G N^* \langle\sigma\rangle_r} \quad (10.16)$$

where $\langle\sigma\rangle_r$ is the effective cross section at the resonance energy for a foil of the thickness used. For a further discussion of the sandwich method see Beckurts and Wirtz (1964).

10.4.4 Thermal Neutron Flux

Thermal neutron fluxes are most easily measured using the cadmium difference technique. Two foils of the same mass are irradiated at the same location, one with cadmium covers to stop the thermal neutrons and one bare. The foils are counted to obtain C_o^{Cd} and C_o^{bare} . The equivalent 2200 m/sec flux is given by

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$$\Phi_{2200} = \frac{C_0^{\text{bare}} - C_0^{\text{Cd}}}{\epsilon G N^* \sigma_0} \quad (10.17)$$

where σ_0 is the cross section at the energy corresponding to 2200 m/sec. If the cross section is $1/v$ below the cadmium cutoff energy, which is often a good approximation, and if the neutron spectrum can be expressed as a maxwellian with temperature T , then we can write a thermal flux and cross section as

$$\Phi_{\text{th}} = \frac{2}{\sqrt{\pi}} \sqrt{\frac{T}{293.6^\circ\text{K}}} \Phi_{2200} = \int_0^\infty \phi(E) dE \quad (10.18)$$

$$\langle \sigma \rangle_{\text{th}} = \frac{\sqrt{\pi}}{2} \sqrt{\frac{293.6^\circ\text{K}}{T}} \sigma_0$$

10.4.5 Cadmium Ratio

A simple way of characterizing the degree of thermalization of a neutron field is to determine the "cadmium ratio." The measurement is the same as that used in the cadmium difference technique described above, but now the quantity $R_{\text{Cd}}(x)$ is defined

$$R_{\text{Cd}}(x) \equiv \frac{C_0^{\text{bare}}}{C_0^{\text{Cd}}} \quad (10.19)$$

where x is the foil material, e.g., Au, Cu, or In. The cadmium ratio is then the ratio of the total neutrons, sub-cadmium plus epi-cadmium, to the epi-cadmium neutrons. Since the cadmium cutoff energy is $E_{\text{Cd}} \sim 0.5$ eV, the ratio tells us nothing about the fast neutrons at a factor of 10^6 higher energy. If we use the simple approximations for the epi-thermal and thermal fluxes and for cross sections given in Equation 10.5, 10.6, and 10.7 we obtain

$$R_{\text{Cd}} = \frac{\int_0^{E_{\text{Cd}}} \sigma(E) \phi(E) dE + \int_{E_{\text{Cd}}}^\infty \sigma(E) \phi(E) dE}{\int_{E_{\text{Cd}}}^\infty \sigma(E) \phi(E) dE} = \frac{\langle \sigma \rangle_{\text{th}} \Phi_{\text{th}} + I \phi_0}{I \phi_0}$$

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$$R_{Cd} - 1 = \frac{\langle \sigma \rangle_{th}}{I} \frac{\Phi_{th}}{\phi_0} \quad (10.20)$$

$$\text{and } \frac{\Phi_{th}}{\phi_0} = \frac{I}{\langle \sigma \rangle_{th}} (R_{Cd} - 1) \quad (10.21)$$

Notice that, since the cadmium ratio depends on the ratio $\langle \sigma \rangle / I$, quoting a value without specifying the foil material is meaningless.

10.4.6 Spectrum Unfolding

If foils of several different materials are irradiated, it is possible to extract a spectrum shape $\phi(E)$ from the combined results. The "unfolding" procedure requires numerical integration and is usually done on a computer. The method is described in IAEA Report 107, and in ASTM Standards E 721 and E 261.

10.5 CORRECTIONS TO DATA

Accurate flux measurement requires that any perturbing effect be accounted for and that the assumptions involved in the data reduction be met. The effects occur either during irradiation or during the counting step, and these will be treated separately.

10.5.1 Irradiation Effects

It has been assumed that the flux is time-independent during the irradiation and that no appreciable "burn-up" occurs of either the target or the product nuclei. If the time dependence of the neutron flux is known, the activation equation can be solved exactly, and this variation taken into account. If the dependence is not known, a bound on the magnitude of the effect can be computed. The magnitude of any burn-up can be calculated using the known cross sections. If the reaction of interest has a threshold above the cadmium cut-off, the use of cadmium covers can minimize the burn-up.

If impurities exist in the foil material the activity to be measured may be produced by more than one reaction. For example, an aluminum foil could be used to determine the fast neutron flux by the reaction $Al(n, \alpha)^{24}Na$. However, elemental sodium from an impurity or from handling would give the same radioactivity by the reaction $Na(n, \gamma)^{24}Na$. Again this effect would be reduced by

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the use of cadmium covers, and a separate measurement in a thermal neutron field could determine the sodium impurity.

Neutron self-shielding and flux depression are two perturbations which result in the apparent flux measured to be less than that which existed in the undisturbed neutron field. Self-shielding refers to the fact that the neutrons are attenuated in the foil and that the activation at the center is less than at the surface. A review of this effect is given by Fleming (1982). Flux depression refers to the fact that the presence of the absorber foil reduces the neutron flux in a medium. Both of these effects are discussed in detail in Beckurts and Wirtz (1964).

10.5.2 Counting Effects

Once the gamma-ray spectrometer has been accurately calibrated for efficiency (see ASTM E 522), the sources of error are generally either count rate dependent effects or gamma-ray absorption in the foil. Count rate dependent effects include dead-time, pulse pile-up, and random summing. These effects are discussed in the references on gamma counting and in De Soete (1972).

10.6 SUMMARY

From this brief introduction to the use of activation foils for neutron field characterization the simplicity and the power of the technique are apparent. Activation foils provide a direct and absolute method for the integral measurement of neutron flux. Although differential measurements of $\phi(E)$ are possible, the results are seldom as direct or unambiguous as can be provided by neutron diffraction or time-of-flight. Even then the absolute normalization of the results might well be done by foil activation.

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CHAPTER 11

CALORIMETRY, CHEMICAL DOSIMETERS, and CERENKOV DETECTORS

G. S. ROESSLER

11.1 CALORIMETRY

11.1.1 Introduction

A calorimeter is used to measure the amount of heat produced in a medium as a result of absorption of ionizing radiation. Because of this, it is a fundamental instrument for absorbed dose measurements. This method for measuring dose deposition does not require evaluation of quantities such as stopping powers, absorption coefficients, and attenuation corrections; therefore, it is one of the few absolute methods of dosimetry. Calorimeters have been employed to measure the energy fluence of x-ray and electron beams as well as to measure the dose produced in the absorption of these radiations. They have also been used to measure the activity of radioactive sources and to calibrate radiation chemical yields. Calorimeters have frequently been used to calibrate instruments to be used as secondary standards.

Calorimetric or microcalorimetric methods have been used in radiation measurements since the experiments of Curie and Labarde in 1903. Calorimetry has been used at the National Bureau of Standards both for the intercomparison of national radium standards and for the standardization of many artificially produced radionuclides and ^{210}Po (NCRP, 1978). Calorimeters are used in the medical field for measurement of high-intensity beams of radiation such as those used in radiation therapy and in the nuclear power field for the assay of plutonium derived from spent nuclear fuel. Calorimeters also have some disadvantages and limitations: 1) although their sensitivity is adequate for radiation intensities and dose rates used in diagnostic, therapeutic, and most industrial applications, it is insufficient for radiation protection purposes; and 2) certain types of radiation processes can lead to a loss of some of the energy that should be measured by the calorimeter, or to a gain of energy from sources other than the radiation field (Laughlin and Genna, 1966). These are referred to in the literature as heat defects, which are usually determined experimentally. Evidence indicates that it is negligible in graphite, a common material used in constructing absorbed dose calorimeters. It is more significant in most plastic materials.

11.1.2 Principles of Operation

Two types of calorimeters that are used at present in radiation dosimetry are shown in Figure 11.1. A small body in which absorbed dose measurements are made (A) is usually of high thermal diffusivity and may be surrounded by nested bodies separated by insulation, such as a vacuum (absorbed dose measurements can also be made in a large single body of low thermal diffusivity, a property which retards a change in temperature profile produced by a beam so that an accurate temperature rise can be made at a point). A temperature measurement in a high thermal diffusivity body assumes that it represents the average temperature throughout the body. This type (depending on the size of the body) is used to measure the total beam energy transported in an exposure (B) or to measure local absorbed dose, while the type consisting of a large single body of low thermal diffusivity is used only to measure local absorbed dose.

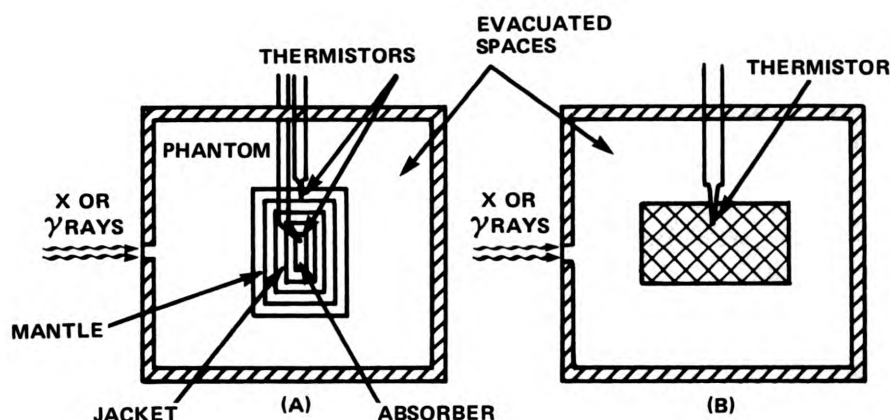


FIGURE 11.1 SCHEMATIC DIAGRAMS OF CALORIMETERS DESIGNED TO MEASURE (A) ENERGY LOCALLY ABSORBED AND (B) TOTAL BEAM ENERGY.

The temperature rise is usually sensed by a micro-bead thermistor placed in an arm of a Wheatstone bridge which is initially balanced by a variable resistor (R) placed in another arm. Radiation causes a temperature rise which decreases the thermistor resistance. This causes an unbalance of the bridge which can be restored by adjusting R so that after beam radiation its percent resistance change (p_b , which includes a heat loss correction) can be determined. A similar quantity (p_c) is determined after electrical calibration where an accurate amount of electrical energy (J) is dissipated in an embedded resistor. In a local absorbed dose calorimeter, where this energy is dissipated in a small isolated mass, m , (in which effects of heat defects and small amounts of foreign materials are negligible),

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the absorbed dose is given by:

$$D = (p_b) \cdot \frac{J}{mp_c} \quad (11.1)$$

If J is in joules and m is in kilograms, D is in gray (symbol Gy). 1 Gy = 100 rad.

In a large body of low thermal diffusivity (assuming a negligible heat defect) the absorbed dose at the position of the thermistor is:

$$D = p_b \left(\frac{^{\circ}C}{p_{\Omega}} \right)_s C_p \quad (11.2)$$

where $(^{\circ}C/p_{\Omega})_s$ is the sensitivity of the thermistor expressed as its temperature change per percent resistance change (p_{Ω}), and C_p is the specific heat capacity of the material at constant pressure. If C_p is in J/(kg $^{\circ}C$), D is in Gy.

11.1.3 Performance

Although calorimeters are regarded as primary instruments in measuring (for example) absorbed dose, they were constructed by a relatively few experimenters. The principle reason is that they are still regarded as uneconomical instruments to use, because long times are thought to be necessary for them to stabilize enough in temperature so that small temperature rise measurements can be made. This major difficulty can be overcome by use of a method described by Domen and Lamperti (1974) who describe a method of quickly restoring a four nested-bodied graphite calorimeter to equilibrium. This calorimeter was used to measure an absorbed dose rate as low as 5 rad/min (which caused a 70 $\mu^{\circ}C$ /min rate of temperature rise), measured to a precision of about 1-2 percent standard deviation of a single run, and to an accuracy within 1 percent.

Measurements of absorbed dose in solid materials are usually converted to absorbed dose to water, which is the primary medium of interest in the medical field. Various uncertainties in the conversion would be circumvented by making direct measurements of temperature rises at a point in a water calorimeter which was widely regarded as impractical. However, its practicability is described by Domen (1982), who describes an efficient absorbed dose water calorimeter. It can be put into operation within an hour. A three minute irradiation produced an absorbed dose of 3.3 Gy (which produces a 0.8 $m^{\circ}C$ temperature rise) that was measured to a precision of 0.6 percent standard deviation of a single run. Irradiations from a ^{60}Co source revealed an exothermic heat defect of 3.5 percent in distilled water. A method was devised to compare these measurements with those determined in polystyrene which is known to have an essentially zero heat defect. This was recently done with a

polystyrene-water calorimeter described by Domen (1983).

The absorbed dose calorimeter has proven to be a practical and portable instrument. In addition to the graphite calorimeter described above, McDonald, et al., (1976) describe a nested bodied A-150 plastic (tissue equivalent) calorimeter that was extensively used as a field instrument.

Measurements of the total beam energy transported in an x-ray exposure is also of fundamental interest. Calorimetric measurements were made with bremsstrahlung beams with maximum photon energies from 6 to 170 MeV (Pruitt and Domen, 1962). The measurements were used to calibrate a described specially designed ionization chamber which was duplicated in various laboratories. Detailed machine-shop drawings of the chamber are included in that report.

11.1.4 Summary

Calorimeters have recently been demonstrated to be practical laboratory and field instruments for measuring absorbed dose. They can be made to be brought relatively quickly into operation, in contrast to a previously time-consuming method of waiting for them to sufficiently stabilize in temperature.

Measurements of the total beam energy transported by bremsstrahlung beams, with maximum photon energies from 6 to 170 MeV, are made with an existing specially designed ionization chamber calibrated with a calorimeter.

Calorimeters have been shown to be useful in medicine, in radiation chemistry, and in the nuclear power field. They are unique in that they themselves can be used to calibrate instruments used as secondary standards. A carefully built calorimeter can have an absolute accuracy within 1 percent.

11.2 CHEMICAL DOSIMETERS

11.2.1 Introduction

The increased use of kilocurie strength radiation sources in many areas of industry and research prompted the need for a type of dosimeter which is reliable for making measurements in high radiation fields. Research on the development of this type of dosimeter is not new. Both solid and liquid dosimeters have been developed in which color change, photo-luminescence or other alteration is a measure of radiation dose. In high dose situations, a dosimeter based on chemical changes produced in an appropriate medium has the potential of fulfilling most desirable dosimeter characteristics. Chemical dosimetry systems, because they usually consist of aqueous solution, have the desirable feature of closely approximating the atomic

composition and density of many biological materials.

However, chemical dosimeters, used primarily for measuring high doses beyond the normal range of ionization chambers, have limited application in routine radiation protection practices where monitoring usually requires a response below 1 rad. In fact, most applications of chemical dosimeters are in the range from 10^4 to 10^5 rads. They are used primarily in situations requiring maximum accuracy and as such are used in some cases for calibrating other dosimetry systems.

11.2.2 Principles of Operation

Theory of Operation

Ionizing radiation initiates chemical reactions in materials; the number of molecules affected is related to the energy absorbed by the material. Ions produced by the radiation combine chemically to form new compounds or change chemical characteristics from those existing in the pre-irradiation stage.

The principle of such dosimetry is based on the Bragg-Gray theory which states that the energy absorbed per unit mass of a substance (dE/dm) is related to the degree of ionization (J_m) per unit mass produced in a small air cavity within the material. Therefore,

$$\frac{dE}{dm} = W S_m J_m \quad (11.3)$$

where W is the energy absorbed per ion pair in the gas and S_m is the relative mass stopping power of the gas (Price, 1964). An important but extremely tedious part of this calculation involves the computation of the average value of S_m . In general, S_m is a function of electron energy, and an average value of it must be calculated for each energy in the spectrum of radiation to which the equation is applied.

Two requirements must be met in order to have an accurate dosimeter. First, it is important for the dosimeter and the sample to have the same mass stopping power; second, the dosimeter and the sample should be geometrically alike. Ionization chambers do not meet these requirements when they are used with liquid and solid samples. However, chemical dosimeters, since they are composed of solids or liquids, do. They not only can have the same mass stopping power as the sample, but can be made into appropriate volumes and shapes.

The response of the dosimeter should be:

1. Proportional to the radiation dose over a wide dose range.
2. Dose-rate independent.
3. Energy and LET independent.

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4. Reproducible (between about 1% and 5%).

It is also desirable for the dosimeter to be stable both before and after irradiation with regard to exposure to light and air and to be able to be used according to a straightforward procedure.

Types of Detectors

Chemical dosimeters can be classified into two groups: 1) the type utilizing water as a solvent, and 2) all other types. Aqueous dosimeters are used if the sample is an aqueous solution, biological material or organic substance.

The effect produced in one of the aqueous types of chemical dosimeters is due primarily to the radiation effect in water. The primary phenomena as a result of the irradiation of water is the formation of the free radicals H and OH. These then may react to form the products hydrogen (H_2) and hydrogen peroxide (H_2O_2). They may also recombine to form water.

The next step in the reaction involves the action of H_2 as a reducing agent on any chemical which is present and capable of being reduced. In a similar manner, H_2O_2 acts as an oxidizing agent on any chemical capable of being oxidized.

These chemical changes are due to excitation and ionization caused by charged particles in the solution. The number and kind of energetic states formed are independent of the type of radiation and degenerate in water, for example, to form chemically active species in about 10^{-12} to 10^{-11} seconds.

By means of chemical titration or absorption spectroscopy, the number of radicals formed in a solution can be measured. The yield of the reactions is expressed in terms of the number of molecules or radicals formed per 100 eV of absorbed energy. This yield is referred to as the "G-value" for any particular chemical system.

Calculation of dose requires a knowledge of the G-value for the reaction which is found by calibration of the particular chemical system by calorimetry and other physical standards. Chemical dosimeters are, therefore, secondary dosimeters. Once the G-value has been determined, absorbed energy can be determined by the relationship:

$$E = \frac{n \cdot 100}{G} \quad (11.4)$$

where E is expressed in electron volts per unit volume, G is, as stated earlier, the number of molecules or ions produced per 100 eV of absorbed energy, and n is the number of molecules formed per unit volume. Chemical titration is commonly used to determine n, although absorption spectroscopy may be more convenient.

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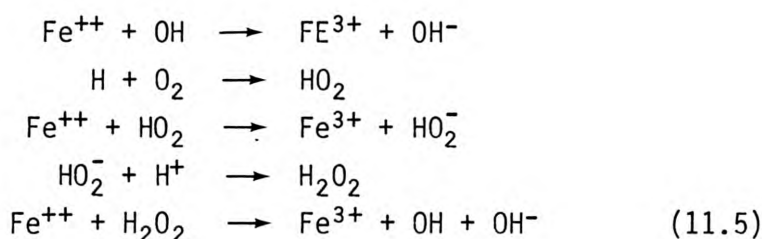
Chloroform-type Dosimeters

A typical chemical decomposition indicator consists of a mixture of water and a chlorinated hydrocarbon such as chloroform, trichloroethylene, or tetrachloroethylene. When this mixture is exposed to radiation it produces hydrochloric acid in proportion to the radiation absorbed. The amount of hydrochloric acid produced may be indicated by means of organic dyes, such as bromocresol purple, which change color depending on the pH. Since small changes in pH near the neutral point can be measured accurately and because of the high yield of this reaction, doses of fractions of rads can be measured. Under irradiation however, they show undesirable chain reactions, poor reproducibility and temperature- and dose-rate dependence.

Ferrous Sulfate Dosimeters

The ferrous sulfate-type dosimeter is the most widely used chemical dosimeter because it is the best understood and has the best dosimetry characteristics. This technique is widely accepted as an absolute standard and is considered reliable with an uncertainty of 1 to 2 percent. It is a method recommended by the Subcommittee on Radiation Dosimetry of the American Association of Physicists in Medicine for quantitating absorbed dose for high energy electrons (SCRAD, 1966). The most useful range of this dosimeter is from 4×10^3 to about 4×10^5 rads.

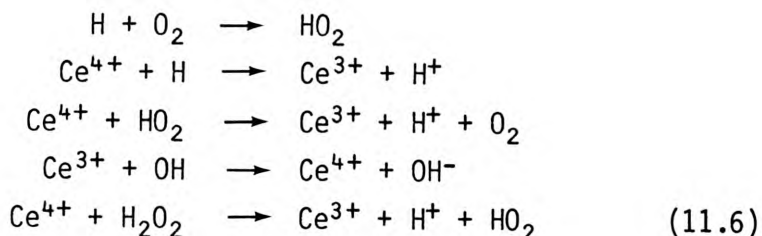
The reaction involved in the ferrous sulfate dosimeter is the oxidation of an acid solution of ferrous sulfate to the ferric salt in the presence of oxygen and under the influence of radiation. The number of ions oxidized is directly related to the absorbed energy due to x-ray or gamma-ray radiation. The simplified reactions are (Eichholz and Poston, 1979):



Ceric Sulfate Dosimeters

The ceric sulfate dosimeter is used to measure doses above those measured by the ferrous sulfate method, from 4×10^4 rads to about 10^8 rads. The chemical reaction involved consists of the reduction of ceric to cerous ions. Equations describing the

mechanism are:



Ceric sulfate solutions must be handled with extreme care while being stored and also during and after irradiation due to their sensitivity to light.

Nonaqueous Dosimeters

Nonaqueous dosimeters which undergo chemical changes are also available. For example, nitrous oxide when irradiated forms N_2 , O_2 , and NO . The nitrogen oxides are condensed and the remaining gas is analyzed. This system is sensitive above 4×10^4 rads. Another dosimeter involves irradiation of monomers or polymers in solution or in pure form in which polymerization, cross-linking, or degradation is induced. Dose determinations are made by measuring the change in viscosity at a constant temperature. Dose ranges are from 10^2 to 10^8 rads.

Other Types of Chemical Dosimeters

Other chemical systems have been developed to cover a wider dose range. These systems, including a ferrous sulfate-cupric sulfate system and an oxalic acid system, are discussed in Eichholz and Poston, 1979. The ferrous sulfate-cupric sulfate dosimeter can be used in the range 4×10^4 to 10^7 rads while the advantage of the oxalic acid system, intended for use in the high-dose range, lies in its high thermal stability before and after irradiation and, in addition, its insensitivity to organic impurities. The latter dosimeter is used primarily for nuclear reactor in-pile measurements since neutron activation is negligible in this system compared to the more commonly used ferrous sulfate and ceric sulfate systems.

11.2.3 Calibration

Experimental Techniques

The most important aspect in the methodology involving chemical dosimeters is purity. Commercially available reagents may

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need repurification and it is recommended that water be triply distilled. Even minute traces of contaminants can produce loss of reproducibility and stability. Also, proper cleaning and handling of glassware and irradiation cells are necessary. Contact between the solution to be irradiated and any organic material such as rubber stoppers or tubing or plastic tubing should be avoided. Glassware can be freed of organic matter by treatment with a strong acid-oxidizing agent mixture such as H_2SO_4 and CrO_3 .

Calibration Procedure

Containers

Chemically-resistant borosilicate glass is recommended to hold the chemical solution. The containers should be as similar as possible in geometry and materials as those employed to hold specimens for irradiation. If the specimen containers are metal or organic, a thinwalled glass liner, 1.0 to 1.2 mm thick, should be used inside the container. The containers should be cleaned before use with a recommended cleaning solution and then rinsed thoroughly with triply distilled water.

Reagents

Solutions for the various types of dosimeters should be prepared according to procedures outlined by the American Chemical Society with attention to purity of reagents, purity of water, and cleanliness of containers and all other materials which come in contact with the solutions. Detailed formulas and methodology for the ferric sulfate, the ceric sulfate, and the ferrous sulfate-cupric sulfate dosimeters are available from the American Society for Testing and Materials (ASTM). Basically, these detail the preparation of dosimetry solutions and standard solutions.

Absorbance Determination

A number of dilutions (6-7) of the standard solutions are made to 100 ml or 1 liter (depending on the type of chemical) into volumetric flasks with a sulfuric acid solution. The optical densities (absorbance) of the diluted solutions are read in a temperature-controlled (25°C) spectrophotometer at the absorption peak with a slit width of 0.5 mm, using quartz cells and either distilled water or other recommended solution as a blank.

A plot is then made of the concentration of ferric or ceric ions versus the absorbance of the diluted solutions. The slope at any one point of the curve obtained is the molar extinction coefficient, as determined by that particular spectrophotometer at that particular density level. For a good optical system, the curve should be a straight line.

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Dosimetry Procedure

The dosimetry containers are filled with freshly prepared dosimetry solutions and placed in the radiation field for an accurately measured period of time. A calibrated large (megacurie) ^{60}Co source is appropriate. After the solutions are removed from the radiation field, the absorbances of the irradiated solutions are read using the spectrometer settings specified above. A portion of the unirradiated solution is then read against the blank soon after the other solutions have been irradiated. This is used as the blank value in the calculation of absorbed dose. The absorbance measurements should be carried out as nearly as possible at a constant temperature.

The absorbed dose in rads is obtained from the change in absorbance by multiplying the conversion factor by G , developed as follows:

G is the number of molecules reacting per 100 eV absorbed; therefore, G molecules = 100 eV or 1 molecule = $100/G$ eV. Using Avogadro's number, 1 mole = $(100/G) (6.02 \times 10^{23}) = (6.02/G) \times 10^{25}$ eV. Since 1 eV = 1.60×10^{-12} erg, 1 micromole/gram = $(9.63/G) \times 10^7$ erg/gram. With the density of the dosimetric solution being 1.02 gram/ml, it follows that 1 micromole/liter = $(9.44/G) \times 10^4$ ergs/gram. Since 1 rad = 100 ergs per g, 1 micromole per liter = $9.44/G$ rads. G is the value determined for any particular chemical system. The equation in SI units would be: 1 micromole per liter = $.0944/G$ gray. A curve of absorbancy reading versus may be plotted.

11.2.4 Summary

In chemical dosimetry, the absorbed dose is determined by measuring a chemical change in an irradiated solution. A number of different solutions can be used depending on the dose range being measured. In general, chemical dosimeters are used in high dose (10^4 to 10^5 rads) situations. If a system is correctly used by expert operators, the reproducibility obtained can be ~ 1 percent (standard error). The system can predict the dose accurately to 2 percent (standard error).

11.3 CERENKOV DETECTORS

11.3.1 Introduction

The Cerenkov effect, named for Cerenkov who was the first to study it in detail (1934), produces the bluish-white light emitted from transparent substances in the vicinity of strong radioactive

sources. This phenomena is familiar to anyone who has observed the bright blue glow in the water surrounding the operating reactor core in a swimming pool reactor. The first to investigate the phenomenon was Mallet (1926) who found the spectrum of light to be continuous and also noted that the effect was different than fluorescence and other luminescent phenomena. After Cerenkov's extensive investigation of the phenomenon, Frank and Tamm (1937) proposed a theory to explain the radiation. Ginsburg (1940) proposed an explanation in terms of quantum mechanics.

11.3.2 Principles of Operation

Cerenkov radiation is light that can be considered to be an electromagnetic shock or bow wave occurring when a particle tranverses a substance with a velocity exceeding that of light in the medium. Fast moving particles can travel in tranparent dielectric media with a velocity (v) which, being smaller than the limiting velocity of light in a vacuum (c), is still greater than its phase velocity c/n in that medium, n being the index of refraction of the medium (Peng, 1975). Under these conditions, the electromagnetic field associated with the particle lags behind, and its interaction with the medium gives rise to the electromagnetic radiation known as Cerenkov radiation. Figure 11.2 is a diagrammatic representation of conditions for emission of Cerenkov radiation. The direction of motion of the particle is shown as I and the resulting light wave is II. BAC and B'A'C' are cross sections of conical surfaces of the emitted light at points A and A', respectively. Consequently, the light is not propagated in all directions but only in a direction perpendicular to the wave front. While the particle with a velocity

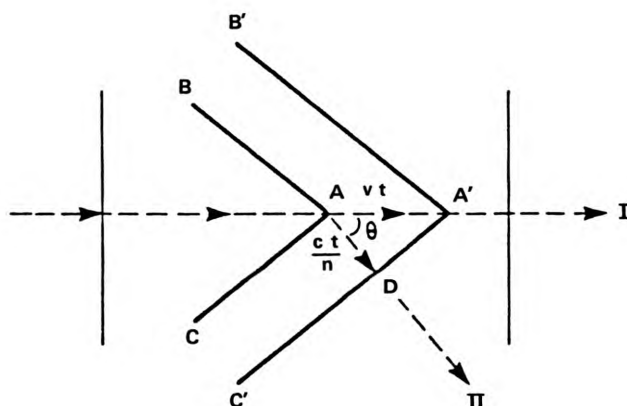


FIGURE 11.2 DIAGRAMMATIC REPRESENTATION OF CONDITIONS FOR EMISSION OF CERENKOV RADIATION. I DEPICTS THE DIRECTION OF MOTION OF THE PARTICLE AND II THAT OF THE LIGHT WAVE. BAC AND B'A'C' REPRESENT CROSS SECTIONS OF CONICAL SURFACES OF THE EMITTED LIGHT AT POINTS A AND A', RESPECTIVELY (PENG, 1975).

(v) travels from A to A' in time (t) which is a distance vt along the direction of motion, the light is propagated with a phase velocity c/n in the same time interval through a distance AA' equal to ct/n . The angle θ between the direction of the motion of the particle and that of the emitted light is:

$$\cos \theta = \frac{ct/n}{vt} = \frac{1}{\beta n} \quad (11.7)$$

since $\beta = v/c$. Cerenkov radiation occurs only when $n > 1$.

Frank and Tamm (1937) derived a practical formula for N_e , the number of photons produced along the track of the electrons after complete dissipation of its kinetic energy in the medium:

$$N_e = 2\pi\alpha \left\{ \frac{1}{\lambda_2} - \frac{1}{\lambda_1} \right\} \int_{\beta=\beta_{\max}}^{1/n} \left(1 - \frac{1}{\beta^2 n^2} \right) d\beta \quad (11.8)$$

where α is the fine structure constant and is equal to $2\pi e^2/hc \sim 1/137$, λ_1 and λ_2 denote the spectral range along the path dl , and integration is between the initial and the cut-off energy.

Cerenkov radiation contains components of all frequencies with energy flux larger toward the violet range. Therefore, in water the light appears bluish-white to the eye.

11.3.3 Unique Characteristics

Applications

Since Cerenkov radiation is produced when a charged particle enters a transparent medium such as water at a velocity greater than the speed of light in the same medium, this phenomenon has been widely used in various fields of high-energy physics for detection of fast moving charged particles and measurement of their velocity.

The application now most commonly used in routine measurements utilizes Cerenkov counting of high energy beta emitters. Routine liquid scintillation counting is an efficient technique for counting of low energy beta emitters such as ^3H ($E_{\max}$ for $\beta^- = 0.018$ MeV) and ^{14}C ($E_{\max}$ for $\beta^- = 0.156$ MeV) and ^{32}P ($E_{\max}$ for $\beta^- = 1.71$ MeV).

Sodium and phosphorus are important biological elements which are often studied by radiotracers. Therefore, ^{24}Na and ^{32}P counting by the Cerenkov technique has found wide application. In particular, the procedure is most useful for ^{32}P since this

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radionuclide is a pure beta emitter (no gammas) and the only means for measuring it is by beta counting. Liquid scintillation counters can be used in this application; however, counting is generally carried out with an aqueous solution of radionuclides. This procedure has the advantage that sample preparation is both simple and economical. Also sample recovery can easily be made.

Commercially available liquid scintillation counters and procedures (see Chapter 6) are used; however, there are four important considerations peculiar to Cerenkov counting - energy discrimination, efficiency enhancement, color quenching, and the absence of conventional scintillation fluors in the system. Due to the nature of Cerenkov radiation, quantitative measurement is not subject to impurity quenching or self-absorption.

Energy Thresholds

The energy threshold effect is solely a function of the refractive index of the Cerenkov generator - solid, liquid, or gas. The result of this effect is that particles below the energy threshold create no photon emission; particles above the threshold do generate photon emission which becomes greater as the initial energy of the particle becomes further separated from the threshold energy (Ross, 1976). Table 11.1 lists some Cerenkov energy thresholds for common solvents.

TABLE 11.1 CERENKOV ENERGY THRESHOLD FOR ELECTRONS IN VARIOUS SOLVENTS.

SOLVENT	n	THRESHOLD (KEV)
ETHYL TRIFLUOROACETATE	1.31	283
WATER	1.33	263
ETHYL ALCOHOL	1.36	243
40% SUCROSE	1.40	219
TOLUENE, 84% SUCROSE	1.50	174
BROMOFORM	1.60	145
METHYLENE IODINE	1.76	111

Efficiency Enhancement

Counting efficiencies are usually low since the threshold energy for Cerenkov emission from electrons in water is 263 keV. The Cerenkov photons have a continuous spectral distribution and a defined direction of emission (the photons are not emitted in all directions).

Since the threshold energy for Cerenkov emission varies

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inversely with the index of refraction of the solvent, various liquid compounds have been tested for Cerenkov counting of low-energy beta emitters. However, the efficiency is quite low in comparison to that of conventional liquid scintillation counting. Efficiency is a combined function of the beta particle energy, sample volume, the nature of the dispersive medium (as described earlier), and the presence of a wavelength shifter. A summary of common beta emitting nuclides, their corresponding efficiencies and solvents is given in Table 11.2 (Peng, 1975).

TABLE 11.2 COUNTING EFFICIENCY OF CERENKOV RADIATION FROM β -EMITTING NUCLIDES (PENG, 1975).

ISOTOPE	EFFICIENCY (%)	SOLVENT
^{14}C	< 1	Aqueous Solution
^{22}Na	12, 16 (dry)	α -Chloronaphthalene
	3.5, 6.7 (ANDA)	Aqueous Solution
^{24}Na	13.2	Aqueous Solution
	18	Aqueous Solution
	25	Aqueous Solution
	19, 52 (dry)	α -Chloronaphthalene
^{36}Cl	1, 12 (dry)	α -Chloronaphthalene
	2.3	Aqueous Solution
	7.7, 23 (ANDA)	Aqueous Solution
^{32}P	10, 55 (dry)	α -Chloronaphthalene
	13	(Dry Sample)
	15	(Moist Sample)
	16	KH_2PO_4 Solution
	25	Aqueous Solution
	30.5	Chloroform/isopropanol
	48	(2/1, v/v)
^{40}K	14	Water (in polyethylene vial)
	27	Aqueous Solution
^{42}K	42, 82 (dry)	Aqueous Solution
	60	α -Chloronaphthalene
	82, 93 (ANDA)	Aqueous Solution
^{47}Ca	7.5	Aqueous Solution
^{56}Mn	40	Aqueous Solution
^{59}Fe	< 1	Aqueous Solution
^{86}Rb	23	Aqueous Solution
	33, 61 (ANDA)	Aqueous Solution
^{90}Y	(50)	Aqueous Solution
$^{90}\text{Sr}-^{90}\text{Y}$	14.2	Aqueous Solution
^{106}Ru	(81)	Aqueous Solution
^{106}Rh	62	Aqueous Solution
^{128}I	40-50, 48-60	Aqueous Solution
	(ANDA)	
^{137}Cs	2.1	Aqueous Solution
$^{144}\text{Ce}-^{144}\text{Pr}$	(75)	Aqueous Solution
	5.4	Aqueous Solution
^{198}Au	5.4	Aqueous Solution
^{204}Tl	1.3	Aqueous Solution

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It has been shown that the use of wavelength shifter can result in a significant increase in counting efficiency. However, caution must be exercised when shifters are used since chemical quenching of the waveshifter can occur.

Baillon et al., (1979) have shown that an increase by about a factor of 3.4 in photoelectron emission from Cerenkov light can be obtained by coating a standard photomultiplier tube with a thin deposit of an organic wavelength shifter. Although this approach apparently did not prove to be feasible for routine counting, it led to the design of a new type of counting vial where a wavelength shifting solution is placed in the outer chamber of the vial consisting of two concentric chambers. The inner chamber is used to hold the sample (Ross, 1976).

Color Quenching

Although chemical quenching does not exist in Cerenkov counting, color quenching can occur. Much of the color quenching problems can be eliminated by the use of strong chemicals to treat the samples. Some users of the Cerenkov technique choose not to use severe color correction methods. The associated disadvantages of time, materials, and expense may not be justified for some samples (Moir, 1971). The channels ratio method (Peng, 1975) is recommended as the best correction method for color quenching.

11.3.4 Calibration

The channels ratio method and the external standardization method have been applied independently or in conjunction with each other (Elrick and Parker, 1968) or with a spectrophotometric method in Cerenkov counting. It has been shown that although the external standardization is useful in Cerenkov counting it is inherently more variable and less accurate than the channels ratio method. This is due to variations in the counting vial which can cause substantial change in the light emissions from the external standard ^{226}Ra .

Therefore, it is recommended that a calibration standard for each radionuclide counted should be prepared by adding a known amount of the standard solution to a sample container of the same geometry as the samples.

All other calibration procedures are as outlined in Chapter 6 for liquid scintillation counting.

11.3.5 Summary

Liquid scintillation counters and techniques have been used effectively for several decades to measure high-energy beta emitters via the Cerenkov method. This approach is the technique of choice for ^{24}Na and ^{32}P , commonly used radiotracers in biologic work,

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since the efficiencies associated with Cerenkov counting of these high energy beta emitters are superior to that of the conventional liquid scintillation procedure.

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CHAPTER 12

RADIATION BIOLOGY AND RADIATION PROTECTION

G. S. ROESSLER

12.1 RADIATION BIOLOGY

Radiation exposure can be divided into two broad categories: (1) external exposure and (2) internal exposure. Considering external exposure, penetrating radiation such as x rays, gamma rays, particles or neutrons originate from a source outside the body and irradiate the body and its tissues. In the case of internal exposure, radioactive materials which made their way into the body, usually by inhalation or ingestion, irradiate various organs or tissues. These radioactive materials may pass through the body or may be deposited in the body indefinitely. While this chapter is concerned primarily with external exposures, biological effects can nevertheless be inferred regardless of the manner in which the dose has been delivered as long as the exposure is expressed in terms of absorbed dose and dose equivalent.

12.1.1 Sources of Information

It has been well documented that exposure to ionizing radiation may produce biological damage. Information about these health effects has been compiled since the turn of the century. Shortly after Roentgen's original paper in 1895 announcing the discovery of x rays, the first case of human injury was reported in the literature (Walsh, 1897). The first case of x-ray induced cancer was reported in the literature as early as 1902 (UNSCEAR, 1977).

Harmful effects caused by large exposures to ionizing radiation turned out to be higher than expected in populations such as early radiologists, miners exposed to airborne radioactivity, persons working in the radium industry, and other special occupational groups. The long-term biological significance of smaller but chronic doses of radiation was not widely studied until relatively recently. As a matter of fact most of the knowledge of the biological effects of ionizing radiation on humans has been accumulated since World War II. Most recently much concern has been expressed for effects caused by low level radiation. Epidemiological and statistical evaluations of human populations exposed to low doses have recently gained considerable attention

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(Mancuso, 1977; Bross, 1979). For instance, a reevaluation of the response to radiation at low doses has recently been completed by the BEIR (Biological Effects of Ionizing Radiation) Committee of the National Academy of Sciences (BEIR, 1980).

Available sources of information on effects of both internal and external radiation exposures consist of insect experiments, animal experiments, and human populations, including patients treated with radiation, patients exposed to diagnostic levels of radiation, Japanese bomb victims, victims of radiation accidents, early radiologists, persons living in areas with high background radiation levels, and radiation workers.

Much of the early information on radiation effects was obtained from experiments on insects such as the *Drosophila* (fruit fly). The *Drosophila* was chosen for study, particularly of genetic (inherited) effects, because its chromosomes are easy to observe in the laboratory and many generations can be observed in a reasonable time. This information has provided much of the biological knowledge to date. However, caution should be used whenever information derived in this manner is extrapolated to humans.

Substantial information has also been derived from experiments with vertebrates. Sublethal, lethal, and supralethal exposures of different types of radiation (x rays, gamma rays, and neutrons) have been delivered to dogs, rats, mice, monkeys, and other vertebrates. In these types of experiments, the doses delivered are well known, and dose-response relationships can be determined accurately. Other advantages of using animals to gather radiobiologic information include the relatively short lifespan of these animals, the ability to control most aspects of the exposure, and the large populations that can be studied. Animal experiments have provided important information, including effects of different doses on sensitive tissues such as blood cells or blood forming organs and on the lining of the intestine. Animals are also useful for studying the effects of internally hazardous radionuclides such as radium and plutonium. However, certain symptoms of radiation sickness (such as nausea, headache, and fatigue) that normally appear in humans following large exposures are more difficult to evaluate in animals. Nevertheless, some of the findings concerning radiation effects on mammals can be extrapolated to humans. However, extrapolation must be done with caution.

Large numbers of cancer patients are treated every year with radiation. Even though the treatment is, in most cases, only partial body and the patients treated cannot be considered normal, the information derived from observation of these treatments has considerable value for the understanding of radiation sickness and the effects of radiation on particular organs and tissues in the body. Advantages of using this population to derive radiation effect information lie in the fact that doses administered are accurately measured and the observations closely followed up.

One of the largest sources of data on radiation effects on humans is the population of survivors of Hiroshima and Nagasaki.

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Thousands of Japanese exposed to the gamma and neutron radiation or the bomb have been studied thoroughly for many years. Even the first and second generations following those exposed have been studied. The interpretation of the findings is complicated by a number of difficulties. The radiation doses could only be estimated through indirect means. Thermal burns, blast effects, and epidemic infections also complicated the interpretations. In addition, it was difficult to obtain an appropriate control population.

Another information source, especially in regard to one-time, large dose exposures are the victims of radiation accidents. Although relatively few persons have been exposed at levels which produce symptoms, this group provides most of the information from which the knowledge concerning acute effects is derived. The REAC/TS (Radiation Emergency Action Center and Training Site) at Oak Ridge has been designated the center for the collection of all information on serious radiation accidents.

Unfortunately since the cases reported deal with accidental exposures, the doses are usually not well known and must be back calculated. A compilation of the information concerning most of the severe radiation accidents in the world including treatment and follow-up of the victims who survived was the subject of an international symposium in 1979 (Hubner and Fry, 1980).

12.1.2 Acute Radiation Syndrome

The onset of the acute radiation syndrome follows an acute, whole-body exposure to ionizing radiation at doses above 100 rads. Although rem, the dose equivalent unit, would appear to be the most appropriate unit to use in the discussion of radiation effects, it cannot be applied in acute, high-dose exposure where H and Q are undefined.

The clinical symptoms which can be expected to appear after exposure are shown chronologically in Table 12.1, for the different dose levels. In general, the initial symptoms come from gastrointestinal distress and manifest themselves as nausea, vomiting and diarrhea. This is often called the NVD syndrome for that reason. The severity of these early symptoms and the time of their onset depend primarily on the dose level and on the sensitivity of the individual exposed. As can be seen from Table 12.1, the exposed individual goes through certain phases over a period of about one month following exposure. The first phase, called the "prodromal syndrome", shows symptoms of gastrointestinal (GI) upset. The cause of the reaction is not completely understood. However, two possible explanations have been offered. One is that the GI symptoms come about for reasons not entirely clear in response to the body's autonomic nervous system. The second explanation is that the body is reacting to the large scale production of toxic substances released by cells as they are destroyed by radiation.

At the lethal and sublethal doses, the exposed individual will

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then go through a period of no readily observable symptoms. This period, which can be compared to that observed in other diseases, is called the "latent period". During the second or third week after exposure the person has a return of abnormal symptoms, which may include a general feeling of ill health, loss of appetite, GI distress, fatigue, drowsiness, loss of hair, and internal bleeding. This stage is called the "manifest illness stage", or the "bone marrow depression period".

TABLE 12.1 CLINICAL SYMPTOMS OF ACUTE RADIATION SICKNESS IN RELATION TO POSTEXPOSURE TIME AND DOSE (ARENA, 1971).

TIME AFTER EXPOSURE	100 TO 250 RAD (SUBLETHAL DOSE)	350 TO 450 RAD (LETHAL DOSE)	650 RAD OR MORE (SUPRALETHAL DOSE)
FIRST WEEK	NAUSEA & OCCASIONAL VOMITING WITHIN HOURS NO DEFINITE SYMPTOMS (LATENT PERIOD)	NAUSEA, VOMITING, PALLOR WITHIN A FEW HOURS NO DEFINITE SYMPTOMS (LATENT PERIOD)	NAUSEA, VOMITING, PALLOR WITHIN A FEW HOURS OR MINUTES SHOCK, UNCONSCIOUSNESS DIARRHEA, ABDOMINAL PAINS & CRAMPS ARTERIAL HYPOTENSION FEVER, SEVERE ERYTHEMA, BURNS OR BLISTERS INSOMNIA, RESTLESSNESS
SECOND WEEK		LOSS OF WEIGHT, GENERAL MALAISE, FATIGUE, STOMATITIS BACTEREMIA, FEVER, ANOREXIA, ABDOMINAL PAINS, SEVERE ERYTHEMA	DEATH CERTAIN WITHIN A FEW HOURS, OR A FEW DAYS, DEPENDING ON DOSE.
THIRD WEEK	GENERAL MALAISE, ANOREXIA, MILD ERYTHEMA, DIARRHEA, FATIGUE, DROWSINESS EPILATION	EPILATION INTERNAL BLEEDING & PETECHIA	
FOURTH WEEK AND LATER	RECOVERY PROBABLE	(MENSTRUAL IRREGULARITIES) 50% CHANCE OF DEATH	

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By the fourth week the person will either have succumbed to the exposure or will recover. In humans, a lethal dose is considered to be about 500 rads with supportive treatment (Wald, 1983). Expressed correctly this is given as LD 50/60 which means this is the dose at which 50% of those exposed will die within 60 days. This is a common unit used with many types of potentially lethal substances and is derived from both animal studies and from cases of accidental exposure of humans and, also from radiation therapy patients receiving nearly total-body doses.

The underlying causes of the problems manifested during the acute radiation syndrome can be summarized by stating that there is damage to the hemopoietic system (the blood forming organs), and at higher doses, to the mucosa (the lining of the small intestine).

The three types of hemopoietic system cells that are of interest are the red blood cells, the white blood cells, and the platelets. The main function of the red cells is the transportation of oxygen to the body cells and the return of carbon dioxide to the lungs for disposal. The essential function of the white cells is that of combating infections. The platelets are instrumental in the blood clotting mechanism. Two subsets of white blood cells are of the most concern: the neutrophils and the lymphocytes. The lymphocytes are one of the most sensitive types of cells in the body and are equally sensitive to radiation at all stages of development, from stem cells to the mature lymphocyte as it appears in the circulating blood. This is in contrast to other types of blood cells which are most sensitive in their early stages of development as stem cells and least sensitive at maturity.

Figure 12.1 shows a typical hemotologic response of the human body to a radiation dose of 300 rads. It is interesting to note that the response of the hemoglobin (red blood cells) to radiation is the least dramatic of the four curves. This is thought to be due to the fact that the red blood cells have a longer lifetime than the other cells, and therefore, few stem cells suffer damage and their loss is spread over a longer time. The lymphocytes drop off precipitously after radiation exposure, due to their high sensitivity even in their mature stages. The platelets drop off somewhat more gradually but reach a level of near zero at one month. The gradual leveling off is thought to be an indirect effect because the radiation resistant platelets that die a natural death are not replaced by their precursors, since they were permanently damaged by radiation. The neutrophils show an unusual increase in numbers shortly after exposure. Their number then drops rapidly for a time followed by a possible increase before falling off to near zero at one month. This unusual pattern of response is thought to be due to a rapid maturation of the neutrophils in their precursor stages in the bone marrow, and is interpreted as a sort of mobilization phenomenon in response of widespread injury.

In cases where the blood count falls drastically and then rises again in about one month, the patient recovers. In cases where there is no recovery, the patient ultimately succumbs due to several

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causes, including the inability to properly clot blood and to combat infection. The seriousness of the hemopoietic injury is a function of dose, and the chance of recovery depends largely on the number of stem cells and other precursors that have survived.

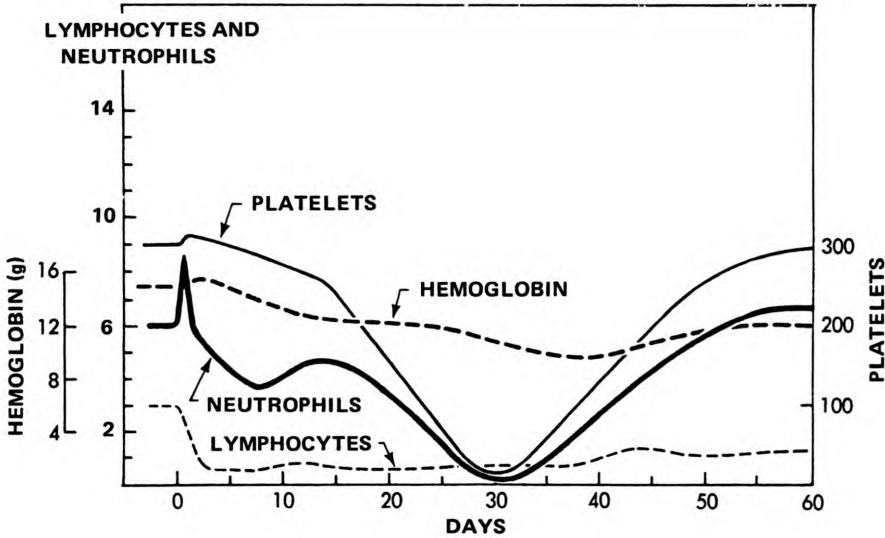


FIGURE 12.1 TYPICAL HEMATOLOGIC RESPONSE OF THE HUMAN BODY TO RADIATION (ARENA, 1971).

Other major problems following radiation exposure concern damage to the mucosa in the small intestine. The mucosa is the most radiosensitive of all the tissues that make up the GI track. In particular, the mucosa that line the small intestine are extremely radiosensitive. The lining of the small intestine, when viewed microscopically, reveals a pattern as diagrammed in Figure 12.2. The villi structures, which can be described as fingerlike projections, allow for a maximum surface area of the intestine optimizing absorption of essential materials from the GI track. The stem cells of the crypt as shown in the figure are particularly radiosensitive. If they are destroyed by radiation, as is the case for doses in the near lethal range, there is no cell renewal along the walls of the villi. The normal process of development of the crypt cells follows through steps of proliferation, maturation and specialization, and sloughing. As a stem cell such as the crypt divides, two cells will be produced. One will remain in the crypt as a stem cell and the other will mature and move up along the side of the villus. When crypt cells are destroyed and no renewal takes place, the cells in the region of cell maturation are not replaced and existing cells must stretch to cover the span, resulting in a flattened villus. Although this decreases the surface area it is not life-threatening as is the eventual production of voids in the lining which expose the more

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delicate underlying tissues. There is also a loss of plasma and various levels of hemorrhaging. The patient's body is open to bacterial infection from microorganisms that normally live as harmless entities in the intestine.

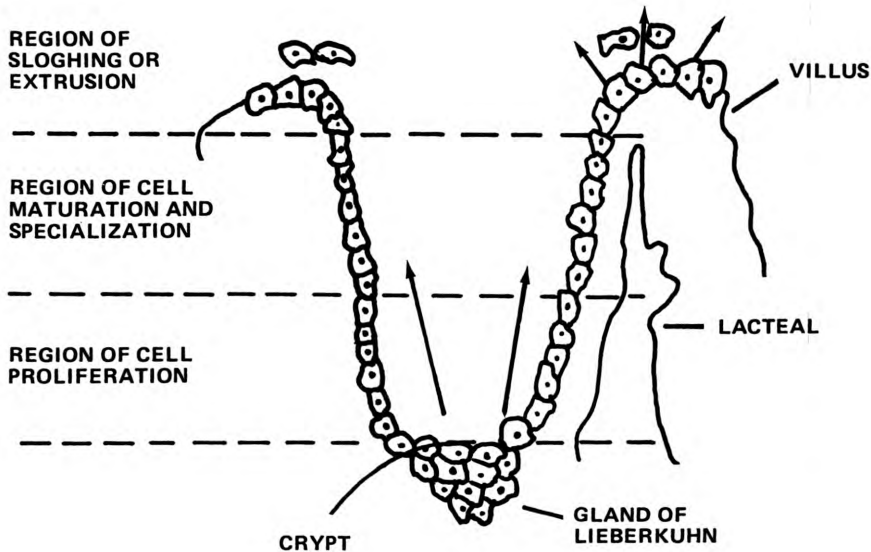


FIGURE 12.2 MICROSCOPIC MORPHOLOGY AND CYTOKINETICS OF THE INTESTINAL EPITHELIUM (ARENA, 1971).

The other tissues of the gastrointestinal system are less radiosensitive than the lining of the small intestine. Nevertheless, their functions may also be impaired by irradiation. Muscular contractions can become abnormal resulting in antiperistalsis and vomiting. Intestinal motility may be upset and altered glandular secretions result in anorexia (loss of appetite), nausea, and diarrhea.

At doses in the supralethal range, the serious damage to the body is thought to be due to the cardiovascular system, or to the central nervous system. Since so few persons have been exposed to these dose levels, the exact cause of death is not known.

In summary, death occurs at lower dose levels (250–500 rads) several weeks after exposure due to effects on the blood forming organs and the inability of the white blood cells to combat infection and the platelets to produce clotting; at intermediate dose levels (500–1200 rads) it occurs in a matter of days and is associated with extensive bloody diarrhea and destruction of the GI mucosa; and at very high doses (10,000–15,000 rads) death occurs in a matter of hours and appears to result from neurological and cardiovascular breakdown.

Treatment of a person exposed to high levels of radiation may fall into three categories. If the dose is less than 400–500 rads the

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patient will probably be watched carefully but will receive no special treatment other than medication for minor symptoms. If the dose exceeds 500 rads, death due to failure of the blood forming organs is likely in approximately one month. These cases require intensive care. Treatment consists of isolation (i.e., keeping the patient from contracting infections from other persons), external sterilization by antiseptic solutions, antibiotics, blood component transfusions, and good nursing practices. Some experts claim that bone marrow transplants should only be used as a last resort, although this may have been the determining factor in saving the lives of four persons who were accidentally exposed to a dose estimated at 700 rads. Other experts recommended proceeding with caution in the area of bone marrow transplantation. A successful bone marrow transplant is used to stimulate the body to begin the production of the essential blood cells again. However, the success of the procedure depends greatly on the familial compatibility of donor and recipient.

12.1.3 Late Effects of Radiation - Somatic

Late effects of radiation may manifest themselves in exposed individuals years after the original exposure. Such delayed radiation effects may result from either previous acute high-dosage exposures, or from chronic low-level exposures over a period of years. Although the acute radiation syndrome is dramatic, it is the long-term effects due to low-level radiation which are more important since the potential exists for damage to a greater number of people.

Most literature describing the long-term effects of ionizing radiation on humans list them as cancer, leukemia, lifespan shortening, embryological defects, and cataracts. Effects of exposure to the unborn child will be dealt with separately in the next section.

Lifespan shortening can be deleted from the list of long-term effects based on the most current information. This applies to the more correctly termed "non-specific life shortening". This is the acceleration of aging due to ionizing radiation. Early research on animals as well as statistics of radiologists' life spans compared to the life spans of other physicians indeed indicated radiation induced aging. However, in recent animal studies which emphasize the use of an appropriate control group, no life span shortening was observed. Ironically, in one experiment on female rats it first appeared that the subjects exposed to low levels of radiation actually lived longer. However, an astute reviewer of the study concluded that the longer life span in this group of rats was due to the fact that they were separated from males during the study and therefore had fewer offspring.

Cataracts is defined as any detectable change of the normally transparent lens of the eyes. Exposure to ionizing radiation can cause cataracts. This has been shown in animal studies as well as in

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analysis of the case histories of radiotherapy patients who received radiation to the lens of the eye. No opacities were observed with doses of less than 200 rads.

Observations made on the survivors of Hiroshima and Nagasaki have been consistent with the data obtained from radiotherapy patients. Only substantial doses of radiation produced vision-impairing cataracts. The available information appears to indicate the existence of a threshold for the induction of detectable lens opacification in humans. The latent period for humans has been reported as from six months to 35 years, with a somewhat inverse relationship between size of dose and time required for cataract development.

Subclinical mental disorders have also been related to radiation exposure (Omran et al., 1978). However, no clear-cut relationship has been identified at this time.

The production of tumors by radiation has been observed since the earliest use of x rays. Radiation carcinogenesis in small animals is well documented. Ionizing radiation has been shown to exert an almost universal carcinogenic action, resulting in tumors in a great variety of organs and tissues. In addition, there is evidence that radiation contributes to the induction of various kinds of neoplastic disease. Some of the experience with humans providing evidence that ionizing radiation is carcinogenic includes 1) the skin cancer frequently observed in radiologists and dermatologists who used radiation in the early years without proper regard for radiation protection (Kathren, 1962), 2) the lung cancer commonly observed in uranium miners who inhale radioactive dust which is deposited in the lung (Waggoner, 1965), 3) the bone tumors found in a large number of dial painters who had the habit of licking paint brushes containing radium and consequently ingesting appreciable quantities of radium (UNSCEAR 1962), 4) cancers of the liver in patients with whom the contrast material thorotrast, which is radioactive, was used to assist in enhancing certain parts of the body for x-rays (Blomberg et al., 1963), and 5) the increased incidence of cancer of the thyroid in children who had been treated with radiation for enlarged thymus glands (Pifer, 1964).

It has also been shown that leukemia, a neoplasm of the blood-forming elements, can be a long-term effect of exposure to radiation. Patients who received radiation for treatment of ankylosing spondylitis, a condition where bones in the spine fuse, show an increased incidence of leukemia (Abbat and Lea, 1958). Also, observations on patients having polycythemia vera, a disease characterized by an overproduction of red blood cells, who were treated with phosphorus showed a significant increase in the chance of developing leukemia years after treatment (Modan and Lilienfeld, 1964). The survivors of Hiroshima and Nagasaki show an elevated incidence of leukemia, as well as solid tumors (UNSCEAR, 1962). Induction of leukemia by radiation stands out because of the natural rarity of the disease, the relative ease of its induction by radiation and its short latent period (BEIR, 1980). In addition,

there are certain types of leukemia which appear to be associated with radiation induction and other types which have never been correlated with radiation exposure.

The latent period for both cancer and leukemia has been fairly well defined. It is usually stated that the latent period for cancer is from 10 to 30 years and that for leukemia it is from three to seven years. In some populations, such as uranium miners, new evidence indicates that cancer may show up in populations exposed to radiation even after 30 years. However, the seven year maximum for development of leukemia is supported by studies on patients having ankylosing spondylitis. In other words, it is unlikely that one would develop radiation induced leukemia after more than seven years following the radiation exposure.

The mechanism of radiation carcinogenesis has been a subject of discussion for many years and at this time there is no universally accepted theory. Two explanations are often offered. The simplest one suggests that carcinogenesis results from the mutation of a normal tissue cell. This is according to one researcher (Hall, 1978). However, this theory is almost certainly too simple to be true. The counter argument is that if all cells in the body suffered the same mutation rate as the germ cells, (for which information is available), then one would expect a vastly greater number of tumors in whole organisms as is in fact observed. An alternative explanation favored by many is that the carcinogenic effect of radiation is indirect. That is, radiation does not create malignant cells itself, but creates conditions which favor the selection of already existing clones of cancer cells which would otherwise either lie dormant or be controlled by normal homeostatic mechanisms.

However, it is agreed that as radiation interacts with human cells the ionization produced in the cell disrupts the chemical bonds on compounds in the cells and consequently disturbs the normal function of the cell. The overall observable effect will depend on the number and the nature of the cells whose function has been altered. These visible effects can range from death caused by cancer or leukemia to genetic mutations.

12.1.4 Effects on the Embryo and Fetus

The immature forms of an animal organism are considerably more radiosensitive than its adult stages, which is in agreement with the famous Law of Bergonie and Tribondeau. This law, which applies for many situations of radiation exposure may be stated in short as follows: "The radiosensitivity of cells is directly proportional to their reproductive activity and inversely proportional to their degree of differentiation."

Figure 12.3 outlines the stages in the development of the human embryo and fetus. As shown, the stages of development include implantation, the embryonic period, and the fetal period. It is during the last two stages that the major organs are formed. Certain

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time periods, shown by the slashed areas, are particularly sensitive to the teratogens which are agents that can induce or increase the incidence of congenital malformations, such as certain chemicals or ionizing radiation.

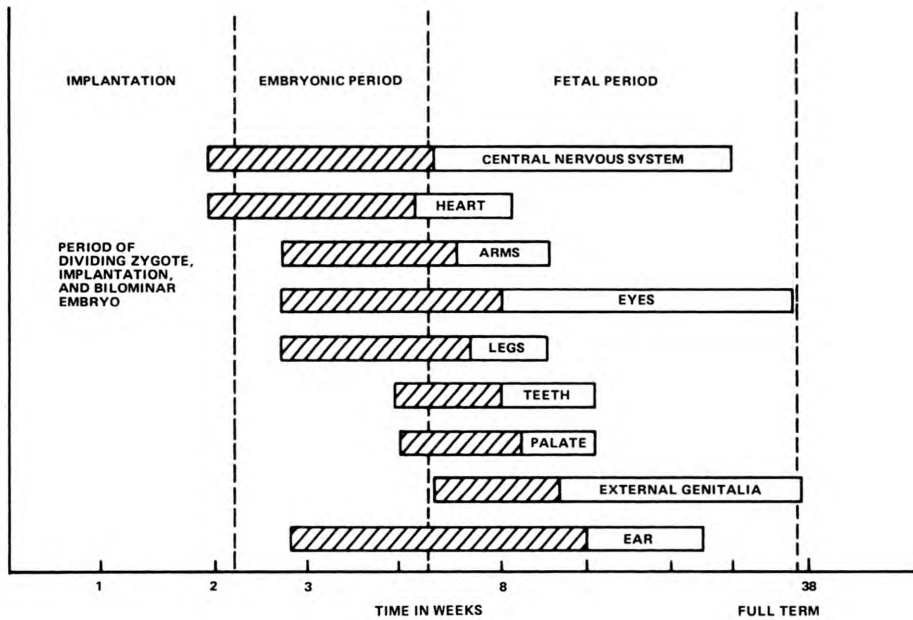


FIGURE 12.3 STAGES OF DEVELOPMENT IN THE HUMAN EMBRYO AND FETUS, SLASHED AREAS REPRESENT HIGHLY SENSITIVE PERIODS.

During the first two weeks of development the embryo is usually not susceptible to teratogens. Depending on the size of the dose, radiation exposure during this stage either damages all or nearly all of the cells, resulting in death, or damages only a few, which allows the embryo to recover without developing defects (Moore, 1977). A corresponding exposure during the fetal period may lead to functional defects and structural abnormalities.

As has been shown in animal experiments, deleterious effects may be produced with doses as low as 10 rem delivered to the embryo. There is no reason to doubt that the human embryo is equally susceptible. In animal experiments during the pre-implantation stage (the first 9-10 days) the principal result of irradiation is a high incidence of prenatal death resulting primarily in a decrease in litter size. During the next month, when the major organs are being formed, radiation exposure causes a higher-than-normal incidence of congenital anomalies (Rugh, 1963). These anomalies are morphological defects such as cleft palate and microcephaly. During the fetal stage it is thought that radiation exposure may produce a higher incidence of functional disorders, which are difficult to evaluate or measure at birth. A certain amount of information is available as a result of irradiation of pregnant women for the

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purpose of therapeutic radium or x-ray treatment. Some information is also available from the survivors of Hiroshima and Nagasaki. A small number of cases of the diagnostic use of x rays in the pelvic area have been evaluated. There is some evidence that doses of a few rads, received in utero from diagnostic x-ray procedures, can subsequently increase the whole spectrum of children's neoplasms, particularly leukemia (Hall, 1978).

Anomalies produced by irradiation of the unborn child are not radiation specific; i.e., they are the same as those which occur in the human population for other reasons. However, there is evidence that they occur at a higher rate.

This very early sensitivity is of great significance because it involves a stage in human embryonic development in which pregnancy may still be unsuspected. For this reason, the International Committee on Radiological Protection (ICRP) has recommended that routine non-emergency diagnostic irradiation involving the pelvic area of women in the child bearing years be limited to the 10-day interval following the onset of menstruation. This precaution would virtually eliminate the possibility of inadvertently exposing a fertilized egg. All of these considerations are gaining importance because of the dramatic increase in the number of females who could potentially be exposed to occupational levels of nuclear radiation, and increased administration of x rays or radionuclides for medical treatment and diagnosis.

12.1.5 Inherited (Genetic) Effects

The single-celled fertilized egg, which results from the union of sperm and egg, contains all of the information necessary to produce all of the organs and tissues of the new individual. This information is carried in the nucleus of the fertilized egg by the 23 pairs of chromosomes, one member of each pair having been contributed by the mother and the other by the father. As each succeeding cell is produced by division, this information is duplicated so that the nucleus in each new cell contains all of the information. Germ cells in the organism will contain the genetic information which eventually contributes to either sperm or egg.

It has been shown that damage can be caused to the chromosomes which will alter the information transmitted to the cells formed by division. Such damage can be produced by a number of agents including chemicals, certain drugs, and ionizing radiation. Chromosomal aberrations have been studied in insects, vertebra, and most recently even in humans, following radiation exposure.

Most geneticists agree that most of these induced mutations are harmful (Neel, 1963). However, if they are damaging they are gradually eliminated from a population by natural means. This is based on the fact that individuals afflicted with this damage are less likely to reproduce than are normal individuals. The more deleterious the mutation, the more likely it is that it will be

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eliminated rapidly. However, mildly damaging mutations may require many generations before they are eliminated completely (Muller, 1954).

Radiation is not unique in its ability to produce mutations and is only one of a number of man-made environmental influences which is capable of increasing the mutation rate. However, for several reasons, it is difficult to measure changes in the normal mutation rate in humans. First of all, the majority of mutations are recessive. Recessive mutations surface only when both parents carry the same genetic damage (UNSCEAR, 1962). Also, generally the damage produced by most mutations is subtle in its effects and difficult to measure. Only if large populations are exposed to the same mutagen does it become possible to observe changes large enough to be distinguished above the natural rate, and only then a link between the mutagen and the effect can be established.

The mutagenic properties of ionizing radiation were first discovered in 1927, using the fruit fly as the experimental subject (UNSCEAR, 1962). Much information had been compiled on mutations in the fruit fly, but since it was a large extrapolation to go from the fruit fly to humans, the scientific community thought it desirable to repeat the genetic experiments with a mammal. It was calculated that the number of animals that would be needed to obtain statistically significant results would approach a million.

The animal chosen for these large scale experiments was the mouse which procreates easily and has a short gestation period. Since it is small, large numbers can be housed and fed more cheaply than larger mammals. It has a considerable number of similarities to humans in structure and function so that extrapolation to humans should be meaningful at least to some extent.

The team of Russell and Russell (Russell, 1963) at Oak Ridge National Laboratory undertook this "megamouse" study and over several decades has provided a mass of information on radiation-produced genetic effects. Among those of health significance are 1) there is no indication of the existence of a threshold dose for the genetic effects of radiation, and 2) the degree of mutational damage seems to be dose-rate dependent, i.e., a given dose is less effective in producing damage if it is protracted or fractionated over a period of time.

Evidence on genetic effects of ionizing radiation in humans is limited primarily due to lack of statistical significance of available data. Populations which have been studied include the Japanese exposed at Hiroshima and Nagasaki and their descendants. Up to the present time, no statistically significant increase in the incidence of genetic effects has been observed (Schull, 1981). However, the various indicators of possible genetic damage studied most recently in the Hiroshima and Nagasaki populations are in the direction expected if an effect was produced.

Recent studies involving humans have shown that ionizing radiation as low as 5-10 rem can cause chromosome aberrations (Dalrymple, 1973). At the present time these studies are limited to

the human lymphocyte (white blood cells) and do not cover germ cells. However, new techniques in human chromosome analysis may contribute significantly in the future to the knowledge of radiation effects causing genetic damage.

The interest in inherited genetic effects stems from the concern that the integrity of the genetic information is maintained. The genetic information is encoded in the DNA molecule. Therefore, much attention has recently been focused on repair of DNA damage resulting from exposure to various environmental agents, including radiation (Hanwalt and Friedberg, 1978). DNA repair mechanisms have been shown to play an important role in the response of humans to radiation exposure. Extensive skin lesions caused by solar radiation exposure are found in individuals having the recessive genetic disease Xeroderma pigmentasum. The cells of these individuals are deficient in a repair system for DNA damage (Cleaver, 1968). Patients having ataxia telangiectasia (Louis-Bar syndrome), a hereditary disorder of males, show a predisposition to cancer and enhanced sensitivity to ionizing radiation (x rays, gamma rays). Cells of these patients have been shown to be defective in repair of DNA damage from x-ray exposure. These observations suggest that repair systems are also involved in the proven ability of humans to resist the onset of cancer following exposure to environmental carcinogens such as solar ultraviolet light and ionizing radiation. It has also been shown that the mammalian cells which are deficient in the DNA repair mechanism also increase chromosome aberrations and can be deleted (e.g., Fanconi's anemia and ataxia telangiectasia) and increased mutation rates are observed (Setlow, 1978).

Meanwhile, a reasonable approach by the medical profession is to recommend to both males and females that conception should be postponed for at least one year following a radiation exposure over 10 R in order to minimize the hazard of inherited effects.

12.1.6 Exposure of the Population to Ionizing Radiation

To put the possible danger of radiation exposure in perspective, an evaluation of existing sources of ionizing radiation and exposures which the population typically suffers is in order. Table 12.2 summarizes the average dose rate in mrem/yr to a typical person in the United States in 1970. Several comments can be made:

1. The doses are well below the minimum rem range in which there are statistically verified effects caused by ionizing radiation.

2. The largest contributor (100 mrem/yr) is the natural environment radioactivity in the soil and building materials, cosmic rays, and the radioactive isotope of the element potassium in our own bodies.

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3. The largest man-made contributor is medical radiation exposure.

TABLE 12.2 SUMMARY OF ESTIMATES OF ANNUAL WHOLE-BODY
DOSE RATES IN THE UNITED STATES (1970) (BEIR, 1972)

SOURCE	AVERAGE DOSE RATE* (MREM/YR)	ANNUAL PERSONS REMS (IN MILLIONS)
ENVIRONMENTAL		
NATURAL	102	20.91
GLOBAL FALLOUT	4	0.82
NUCLEAR POWER	0.003	0.0007
SUBTOTAL:	106	21.73
MEDICAL		
DIAGNOSTIC	72**	14.8
RADIOPHARMACEUTICALS	1	0.2
SUBTOTAL:	73	15.0
OCCUPATIONAL	0.8	0.16
MISCELLANEOUS	2	0.50
TOTAL:	182	37.4

NOTE: * THE NUMBERS SHOWN ARE AVERAGE VALUES ONLY. FOR GIVEN SEGMENTS OF THE POPULATION, DOSE RATES CONSIDERABLY GREATER THAN THESE MAY BE EXPERIENCED.

** BASED ON THE ABDOMINAL DOSE.

12.1.7 Dose Effect Models

Most of the information which relates dose and effect comes from relatively high dose situations. This high level of radiation exposure is not applicable to the situations of occupational exposure or exposure of the public. Especially it is not the dose range of interest for the development of guidelines and regulations concerning persons employed in the nuclear industry or in the area of x-ray diagnostics.

In these cases one needs to have a dose-effect relationship that holds in the low rem and even millirem range. At present the only way to obtain such a relationship is to extrapolate from higher dose levels in humans as well as to use information from animal experiments. For such an extrapolation, a number of biological

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models have been developed (Brown, 1976). One of them is the linear-quadratic model which describes the effect as the degree of incidence of death, cancer, or some other known radiation-induced effect. This model, shown in Figure 12.4, is described by the following equation:

$$I = C + \alpha D + \beta D^2 \quad (12.1)$$

where I is the incidence of the particular effect, D is the dose, C is the intercept of the effect axis, α and β are the numerical coefficients of D and D^2 respectively. The quadratic characteristic causes the effects at lower doses to be not as significant as those at higher doses.

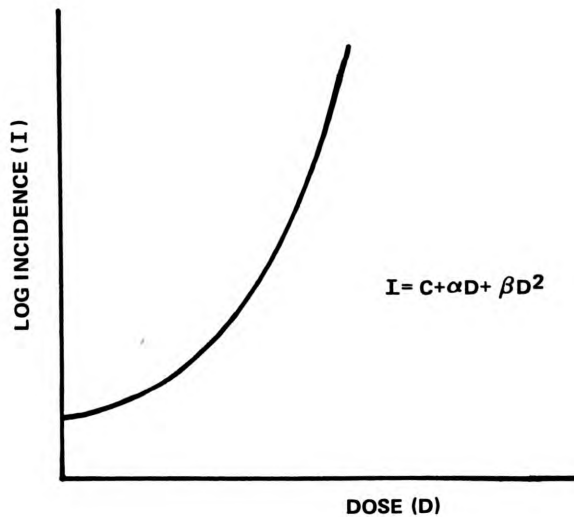


FIGURE 12.4 DOSE-RESPONSE MODEL: LINEAR - QUADRATIC

Recent data on cell survival curves and research concerning induction of mutations in neurospora support this type of dose response curve (Webber, 1965; Douglas, 1975). However, if this model is indeed valid, the question arises as to the point at which the curve departs from linearity. It is the outcome of this question which determines whether the risk at low doses is overestimated.

Another proposed model is shown in Figure 12.5. The analytical description is:

$$I = (C + \alpha_1 D + \beta D^2) e^{(-\gamma D - \delta D^2)} \quad (12.2)$$

where γ and δ are the numerical coefficients of D and D^2 in the exponential term. This model still assumes a linear-quadratic relationship but also includes an exponential parameter which causes

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the model to show a reduced effect at high doses. The reasoning is that if an appreciable number of cells are already killed, the effect per unit dose becomes less.

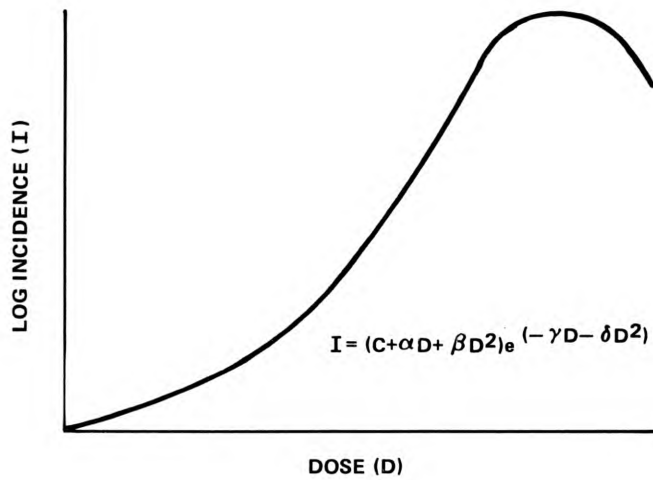


FIGURE 12.5 DOSE-RESPONSE MODEL: LINEAR-QUADRATIC WITH "CELL-KILLING EFFECT".

It should be pointed out that if the model is written incorporating an additive constant (C) a non-zero intercept of the y-axis is experienced. At zero dose there is of course no effect and the model therefore needs to be written as $I = \alpha D + \beta D^2$.

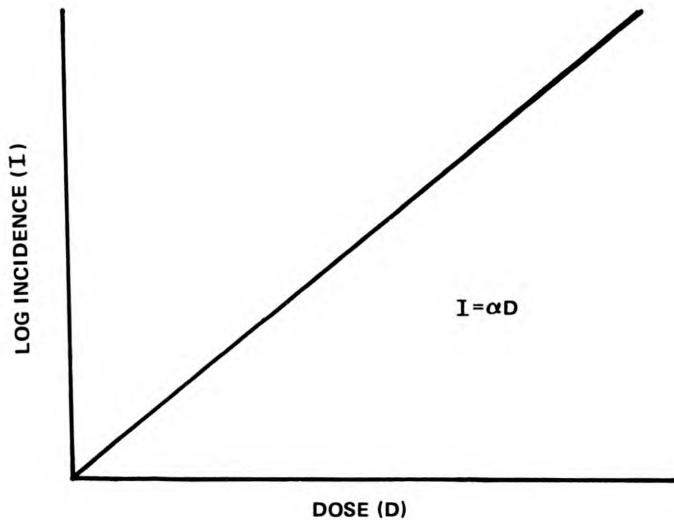


FIGURE 12.6 DOSE-RESPONSE MODEL: LINEAR

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Animal experiments and limited human information conform at high doses to a model described by $I = C + \alpha D$, shown in Figure 12.6. Since a zero intercept is required, the model is usually written as $I = \alpha D$.

This model produces the required zero intercept, as well as the constant slope of the curve which is required if a linear relationship between dose and effect exists. This is the popular linear, non-threshold model that is at present assumed for all radiation effects concerning humans. It serves as the basis for all radiation protection guidelines and regulations.

A recent release of the National Academy of Sciences Committee (BEIR, 1980) which updates the report of 1972 concentrates primarily on long-term somatic and genetic risks to people exposed to ionizing radiation at low doses. Since this area of radiation effects is of principal concern with respect to risk to large population groups, this report may have significant impact on development of future radiation guidelines and regulatory limits. The Committee recognized that there is "great uncertainty in regard to the shape of the dose-response curve for cancer induction by radiation, especially at low doses." This report differs substantially from the 1972 report that in estimating the cancer risk from low doses of low-LET radiation, a linear quadratic dose-response model (as in Figure 12.4) is thought to be more consistent with epidemiologic and radiobiologic data than the more "extreme" dose-response models such as the linear (Figure 12.6) or the pure quadratic ($I = \alpha D^2$, not shown in the figures). The Committee concluded that for exposure to low-LET (LET: Linear Energy Transfer, amount of energy dissipated by an ionizing particle per micrometer of path keV/m) radiation at low doses, the linear model "probably leads to overestimates of the risk of most cancers, but can be used to define upper limits of risk." The Committee suggests using the quadratic model to define the lower risks from such radiation.

The report also concluded that it is not known whether dose rates of gamma or x rays of 100 mrad/year are detrimental to man. Any somatic effects at these dose rates would be masked by environmental or other factors that produce the same types of health effects as does ionizing radiation. Also, the Committee adds, it is unlikely that carcinogenic effects of doses of low-LET radiation at dose rates on the order of 100 mrem/year will be demonstrable in the near future.

12.1.8 Risk Models

In recent years substantial attention has been devoted to the development of such risk models. The ultimate goal is to quantitate the risk of exposure to ionizing radiation in terms of potential health effects. To have such a quantization of the risk is important for management of radiation workers for legal aspects, etc. Various risk models have been developed from irradiated populations such as

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the survivors of Hiroshima and Nagasaki, the British ankylosing spondylitis patients treated with x rays and medical, occupational, and accident groups exposed to ionizing radiation. By now a number of risk models exist and have been summarized (BEIR, 1972; ICRP, 1977). To utilize these risk models for determination of effects in a specified population, a collective dose in person-rem is calculated. This is done by multiplying the number of persons in the population with the estimated average dose for that group.

Present estimates predict that one million person-rem could cause between 87 and 458 fatal cancers. A figure of 200 fatal cancers per one million person-rem is now accepted as a sufficiently conservative model (Ad Hoc Group, 1979). The estimates of inherited effects are less certain due to the lack of evidence of inherited human abnormalities following radiation exposure. However, the estimated risk of inherited abnormalities to all future generations has been projected to be between 36 and 900 effects per one million person-rem (Ad Hoc Group, 1979). This number refers primarily to dominant diseases. The chromosomal and recessive disease effects are thought to be relatively slight. A geometric mean of 180, rounded off to 200, is accepted by the scientific community as the inherited risk model. A similar risk estimate has been generated for non-fatal cancers and is also approximately 200 effects per one million person-rem (Ad Hoc Group, 1979).

Concerning the applications for these risk models, quite a few can be seen. One application might be evaluating the health effects due to the emissions occurring during normal operations of a nuclear power plant. Various scientific groups determined that a reasonable estimate of the maximum dose to an individual does not exceed 15 mrem/yr (Interagency Task Force, 1979). Using the appropriate risk model, this translates to an increased annual chance of death due to cancer of 3×10^{-6} per person. Over a 70 year lifetime, this increase in chance of death due to cancer becomes about two in ten thousand or 2×10^{-4} .

General statements on population dose from a typical individual nuclear power plant must be made with caution. Collective population dose in the vicinity of a specific plant is dependent on many variables including the type and design of plant, the size and distribution with direction and distance of the surrounding population, the specifics of the several release pathways, and the vagaries of the various potential dose pathways in the environment. In addition, since predicting effects based on collective doses presumes a linear model, most evidence indicates that an overestimation will result.

A reasonable generic upper collective dose limit for a nuclear power plant in routine operation might be on the order of 100 person-rem/yr for a population of one million within a 50 mile radius (Roessler, 1979). For these conditions the risk model predicts a theoretical steady-state cancer fatality rate of 0.02/yr (in other words one every 50 years) or between one and two fatalities during the 70-year lifetime of the entire one million population.

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Another application of risk models consists of evaluating health effects to radiation workers. Of interest is the maximum exposed individual, i.e., the individual who actually receives the maximum permitted dose of 5 rem/yr. Applying the risk model under the assumption of 200 fatal cancers per one million person-rem suggests that this individual's increased cancer risk associated with one year of exposure at the maximum permissible level is on the order of 0.001 or one in a thousand. By comparison, a person's lifetime risk of fatal cancer from all other causes is 0.15 or about one in seven (ACS, 1979).

Another value of interest is the aggregate risk to the population of nuclear power plant workers. Existing monitoring data show that of the approximately 72,000 persons monitored in 1977, 44,233 individuals had measurable radiation doses and the average dose being 740 mrem (ITF, Work Group on Exposure Reduction, 1979). If the 1977 levels are taken as typical and if it is assumed that this group of workers was engaged in radiation work for 20 years, a collective dose of 654,000 person-rem is obtained. Applying the risk model, 130 excess cancers during the lifetimes of these individuals are predicted as the upper limit from 20 years exposure while operating under the present standard of 5 rem/yr. This represents an approximately 1 increase in risk over the 10,800 cancer fatalities expected from other causes over the lifetimes of 72,000 persons (ACS, 1979).

12.2 RADIATION PROTECTION

The goal of any radiation protection effort should be patterned after the primary objective of the Health Physics Society which is "... the development of scientific knowledge and practical means for the protection of man and his environment from the harmful effects of radiation, thus providing for its utilization for the benefit of mankind."

Therefore successful performance of radiation measurement involves both aspects of this Society's objective. For radiation instrumentation to be useful, it must be designed, built and operated in a way to satisfy both requirements.

12.2.1 Radiation Protection Guides

Guidelines for radiation protection are based on two basic assumptions. The first is the assumption that radiation effects are a linear, non-threshold function of dose. The non-threshold relationship implies that no radiation protection guideline, no matter how it may be set, can insure absolute safety to any exposed individual. The second assumption is that radiation, like many other developments of technology, has a large benefit potential.

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Consequently, consideration must be given to the extent of the benefits which will make a certain degree of risk acceptable.

The basic philosophy concerning the use of radiation is to maximize the benefits gained by utilization of ionizing radiation, while assuring at the same time that necessary exposures are kept as low as is reasonably achievable (ALARA). The principle of ALARA is an extension of an original recommendation of the National Council on Radiation Protection and Measurements (NCRP, 1954). Similar recommendations are in NCRP reports up to the present time. The basic radiation protection philosophy of these recommendations has been incorporated in regulations and guides of the Nuclear Regulatory Commission (NRC).

Management organization and administration for ALARA is essential for the success of any radiation protection program (Kathren, 1982). Guidelines for achieving this goal and a useful reference list are given in Kathren, 1982.

Radiation guidelines may apply to radiation workers or to the general population. Those for the general population are of equal concern as those for radiation workers. They serve as a basis for many of the considerations applicable to the design and implementation of Environmental Surveillance Programs.

Agencies and Their Philosophies

A variety of different types of guidance concerning radiation protection is provided by numerous groups having different jurisdictions and missions. This radiation protection guidance can take a variety of forms including numerical limits, performance standards, guides to good practice, and review and comment on levels, trends and effects. Numerical limits might be placed on radiation dose received by people, quantities of radioactivity, radiation emissions, radioactivity effluents or ambient levels of radiation and radioactivity. Standards are imposed to insure safe design and operation. These may include a numerical emission limit, require a protection device (such as beam size limitations or interlocks), specify accuracies and reproducibilities (of devices such as exposure timers), or provide other performance or feature specifications. Guides to good practice might contain guidance such as recommendations for surveying and monitoring.

Still another form of guidance is practiced when radiation and radioactivity level limits and standards are periodically reviewed, as they are evaluated according to the latest knowledge of biological effects, and as possible consequences are estimated.

Radiation protection guidance can also be identified according to mission and authority of the entity issuing the guidance. These entities can be roughly classed as 1) scientific and advisory, 2) consensus, 3) legal and regulatory, and 4) operational.

To appreciate the scope of the various organizations either involved in the development of guides or having regulatory responsibility, these four entities are outlined as follows:

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1. Scientific and Advisory: Historically, scientists, physicians, and other professionals acting through voluntary, non-government organizations have reviewed the data available and have issued recommendations on principles and philosophies of radiation protection, numerical limits, and methods of good practice.

These organizations are at both the international and national level. They have no direct legal jurisdiction, but enjoy a considerable degree of respect. These recommendations commonly form the basis for regulations of official agencies.

2. Consensus: This method promotes standardization necessary to industry and commerce. Representatives of users (manufacturers, government agencies, professional societies) meet and adopt consensus standards which are voluntary, but may be adopted by referendum.

3. Legal, Regulatory: This class is drawn up by official agencies based on the recommendation of the advisory groups. Additional bases may be developed where necessary. They introduce legal and regulatory requirements and promulgate regulations which have the force of law.

While many feel the development of radiation guidance should progress independently of regulatory activities, the criteria with which radiation users must ultimately comply are embodied in regulations, and thus, it is virtually impossible to separate radiation protection guides and regulatory processes.

4. Operational: Guidelines may be promoted by the user. Companies, universities, and laboratories for example, draw up work rules for the specific organization or site in an attempt to describe special conditions that will assure compliance.

The agencies and organizations which function in these areas may appear numerous and perhaps overlapping to the uninitiated. For that reason an attempt is made in the following summarization to address all the agencies and organizations according to the outline above, and according to their historical appearance.

1) Scientific and Advisory

INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION (ICRP) - In 1928, the Second International Congress of Radiology established an International x-ray and Radium Protection Commission. The interest of this group expanded beyond the medical and in 1950 the name was changed to the International Commission on Radiological Protection. The ICRP recommends basic principles and basic numerical standards on the international level. National recommendations in turn are usually based on ICRP recommendations.

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INTERNATIONAL COMMISSION ON RADIOLOGICAL UNITS AND MEASUREMENTS (ICRU) - The ICRU was founded in 1925 and works closely with ICRP. The ICRU develops recommendations concerning quantities and units of radiation and radioactivity, procedures for measurements and application of these quantities in radiobiology and clinical radiology, and physical data needed in the application of these procedures.

THE INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA) - The IAEA is the United Nations agency organized in 1956 for the promotion of peaceful use of atomic energy. The agency prescribes health and safety measures for its own operations and for those receiving IAEA assistance. It has issued a series of guides to good practice in its Safety Series.

THE UNITED NATIONS SCIENTIFIC COMMITTEE OF EFFECTS OF ATOMIC RADIATION (UNSCEAR) - This Committee issues periodic reports of world-wide data and projections and also introduces some dose assessment methodology.

Numerous other international organizations such as the International Labor Organization and the International Standards Organization also have some role in radiation protection guidance.

NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENTS (NCRP) - In 1929, an advisory committee on x-ray and radium protection was organized in the United States. Later, this became the National Committee on Radiation Protection and Measurements and it is now the National Council on Radiation Protection and Measurements. This group is independent of any government agency, but does have a federal charter. The NCRP has subcommittees in various areas. Recommendations of the NCRP are almost universally recognized in this country as the best authority on radiation protection. NCRP recommends numerical limits for radiation exposure and also issues recommendations of good practice. The NCRP has published over 60 reports. NCRP reports were originally published as National Bureau of Standards handbooks and printed by the government printing office. This practice helped NCRP economically, but led to the misconception that NCRP reports were recommendations of an agency of the United States Government. NCRP reports are now published independently and sold directly by NCRP. U.S. Representatives to ICRP are NCRP members and the recommendations of the two agencies parallel each other closely.

NATIONAL ACADEMY OF SCIENCES (NAS) - When requested by other federal agencies, the National Academy of Sciences organizes committees to evaluate biological effects data and render opinions on the effects of current radiation limits. These committees also identify areas needing further consideration in radiation limit setting. The most recent NAS reports are those of the Biological Effects of Ionizing Radiation Committee (BEIR, 1972 and 1980).

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FEDERAL RADIATION COUNCIL (FRC)/ENVIRONMENTAL PROTECTION AGENCY (EPA) - The now defunct FRC was actually a transitional organization bridging the gap between the scientific and legal. It was formed in 1959 by executive order and made statutory by Congress in Public Law 86-373. The FRC was assigned the responsibility of advising the government on federal policy, advising the President on radiation protection policy and formulating the basis for recommendations of various federal agencies. The FRC was formed in recognition of the fact that while basic radiation protection guidance was being developed by the prestigious NCRP, there was actually no agency in government assigned the responsibility for developing national radiation protection policy. The FRC drew heavily on the scientific recommendations of the NCRP but it was also charged with weighing nonscientific aspects such as social, political, military, and economic in the risk-benefit evaluation.

With the formation of the Environmental Protection Agency (EPA), the FRC was abolished as a separate agency and the functions were transferred to EPA where they continue to reside.

FEDERAL RADIATION POLICY COUNCIL (FRPC) - In 1979 a report of a review committee appointed by the President concluded that there was insufficient coordination between Federal agencies involved in radiation protection and recommended several possible courses of action to correct this situation. As a result, the Federal Radiation Policy Council was established in February 21, 1980.

The Council was composed of 13 assistant secretaries or equivalent ranking officials from the following agencies - Department of Commerce, Department of Defense, Department of Energy, Environmental Protection Agency, Federal Emergency Management Agency, Department of Health and Human Services, Department of Housing and Urban Development, Department of Justice, Department of Labor, National Science Foundation, Nuclear Regulatory Commission, Department of Transportation and Veterans Administration.

The Council was chaired by the Administrator of the Environmental Protection Agency. The permanent staff and administration support for the Council is provided by EPA.

The Office of Management and Budget, the Office of Science and Technology Policy, the Council on Environment Quality, the Regulatory Council, the Domestic Policy Staff, the Intergovernmental Affairs Office and the Interagency Radiation Research Committee participate as observers in the Council's activities. The Council was abolished in 1981 due to lack of funding.

2) Consensus Organizations

AMERICAN NATIONAL STANDARDS INSTITUTE (ANSI) - The primary organization in the United States for promoting the development of consensus standards (standards representing the consensus opinion of users, manufacturers, and professionals in this specific area) is the American National Standard Institute (ANSI). ANSI develops

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standards in a variety of areas, including the nuclear area. There are a number of ANSI committees in the nuclear area, including the N-13 committee which was charged with developing radiation protection standards (Health Physics Society Handbook, 1978). Under this committee, in turn, are numerous subcommittees working on various aspects of radiation protection and monitoring. ANSI standards, once accepted, become operational guidelines for radiation users in the country and are even adopted by reference in the regulations of agencies.

3) Regulatory Agencies

The following is a selection of the agencies with the greatest degree of participation in radiation regulation:

NUCLEAR REGULATORY COMMISSION (NRC) - The NRC was formed in 1975 and is a successor to the regulatory portion of the former United States Atomic Energy Commission (USAEC). The NRC (as USAEC before it) is limited by its enabling legislation to jurisdiction over what is known as special nuclear, source and by-product material. Special nuclear material is a material that is capable of supporting a self-sustaining nuclear reaction. Special nuclear materials are uranium enriched in the isotopes of U-235 and U-233, and also plutonium. Source material is material from which nuclear material may be derived and includes natural uranium and natural thorium. By-product material is any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material. This includes fission products and radioactive products activated by neutrons from a nuclear reactor. By-product material was defined by Congress in 1980 to include uranium mill tailings. Radiation protection standards are contained in Part 20 or Title 10 of the Code of Federal Regulations (10CFR20).

DEPARTMENT OF HEALTH and HUMAN SERVICES, FOOD AND DRUG ADMINISTRATION (DHHS-FDA) - Several bureaus are involved in radiation regulations. The Bureau of Radiological Health (BRH) derives its authority from Public Law 90-602, the "Radiation Protection for Health and Safety Act of 1968." This act authorizes BRH to develop standards applicable to the manufacture and assembly of electronic products which produce radiation. Standards developed under this legislation are found in Title 21 of the Code of Federal Regulations in parts numbered 1000 and higher (21 CFR-1000's). Under this authority, BRH has developed standards applicable to the manufacture and assembly of diagnostic x-ray machines, microwave ovens, cabinet x-ray machines and lasers.

ENVIRONMENTAL PROTECTION AGENCY (EPA) - In addition to its responsibility for developing general standards for use by various federal agencies, including the armed forces, the EPA also has a

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responsibility for specific regulatory programs. These include the development of drinking water standards under the Safe Drinking Water Act of 1974 and issuance of discharge permits for potential sources of pollution. Under these authorizations, interim regulations have been drawn up for the levels of radionuclides in drinking water and radionuclide limitations have been considered in some discharge permitting activities.

DEPARTMENT OF TRANSPORTATION (DOT) - The DOT has authority for regulating the various modes of transportation of hazardous materials and has regulations for the packaging and labeling of radioactive materials.

DEPARTMENT OF LABOR - OCCUPATIONAL SAFETY AND HEALTH ADMINISTRATION (DOL and OSHA) - OSHA, in its regulations pertaining to occupational hazards, issues specific regulations for ionizing radiation which are however essentially identical to those of the NRC.

STATE AND LOCAL AGENCIES - Since NRC authority is specifically limited to the categories of source, special nuclear and by-product material, states may exercise authority over users of other radiation sources such as radiation producing machines, natural radioactivity other than source material, and accelerator produced radioactive materials. In addition, about half of the states are "agreement states" and thus also have assumed responsibility for by-product source and special nuclear material in quantities not sufficient to form a critical mass. Thus, in non-agreement states, the NRC regulates by-product, source, and special nuclear material while the state has the authority to regulate all other sources. However, in agreement states, the state regulates all radiation sources except critical mass quantities of special nuclear material (in such facilities as fuel fabrication plants, nuclear reactors, and fuel reprocessing plants) for which the NRC retains jurisdiction.

State agencies responsible for regulating radiation sources vary from state to state. Responsibility rests with the state health agency in a majority of the states, but also may lie with state environmental agencies, state labor agencies and atomic or nuclear energy authorities. The City of New York is a co-participant with New York State in the agreement program and has its own regulatory program and regulations. In other states, cities or counties may have agreements within the state and share the work load on a local basis but operate under a single set of statewide regulations and standards. State environmental agencies are also showing increasing degrees of activity in developing environmental radioactivity standards.

4) Operational

Specifics on guidelines at the operational level will vary

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considerably from user to user. Therefore further detail will not be given here.

TABLE 12.3 NCRP DOSE LIMITING RECOMMENDATIONS

A. MAXIMUM PERMISSIBLE DOSE EQUIVALENT -- OCCUPATIONAL EXPOSURE	
1. COMBINED WHOLE BODY OCCUPATIONAL EXPOSURE.	
a. PROSPECTIVE ANNUAL LIMIT	5 REMS IN ANY ONE YEAR
b. RETROSPECTIVE ANNUAL LIMIT	10-15 REMS IN ANY ONE YEAR
c. LONG TERM ACCUMULATION TO AGE N YEARS	(N-18) X 5 REMS*
2. SKIN	15 REMS IN ANY ONE YEAR
3. HANDS	75 REMS IN ANY ONE YEAR (25/QTR)
4. FOREARMS	30 REMS IN ANY ONE YEAR (10/QTR)
5. OTHER ORGANS, TISSUES & ORGAN SYSTEMS	15 REMS IN ANY ONE YEAR (5/QTR)
6. FERTILE WOMEN (WITH RESPECT TO FETUS)	0.5 REM IN GESTATION PERIOD
B. DOSE LIMITS FOR THE PUBLIC, OR OCCASIONALLY EXPOSED INDIVIDUALS	
1. INDIVIDUAL OR OCCASIONAL	0.5 REMS IN ANY ONE YEAR
2. STUDENTS	0.1 REMS IN ANY ONE YEAR
3. AVERAGE OF A POPULATION GROUP	0.17 REM AVERAGE PER YEAR
C. EMERGENCY DOSE LIMITS	
1. LIFE SAVING	
a. INDIVIDUAL (OLDER THAN 45 YEARS IS POSSIBLE)	100 REMS
b. HANDS AND FOREARMS	200 REMS, ADDITIONAL (300 REMS TOTAL)
2. LESS URGENT	
a. INDIVIDUAL	25 REMS
b. HANDS AND FOREARMS	100 REMS, TOTAL
D. FAMILY OF RADIOACTIVE PATIENTS	
1. INDIVIDUAL (UNDER AGE OF 45)	0.5 REMS IN ANY ONE YEAR
2. INDIVIDUAL (OVER AGE 45)	5 REMS IN ANY ONE YEAR

* IT IS EXPECTED THAT THE 5 (N-18) RULE WILL SOON BE ABOLISHED.

Numerical Limits

The philosophy and basis for the setting of radiation protection standards is complicated by the fact that the effects in

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the population caused by low level radiation are identical to those caused by other influences. Also, the form of the dose response curve at low levels and the existence of a threshold for radiation effects is difficult if not impossible to establish.

As stated earlier, standard setting groups assume a linear, non-threshold relationship between dose and effect in the low dose regions for which direct observational data are not available. This does not necessarily constitute a belief in the validity of the linear, non-threshold hypothesis, nor does it provide a basis for extrapolation for accurate prediction of radiation effects, but it does provide a hopefully conservative basis for the setting of standards.

A distinction is also made between exposure received by a radiation worker and by a member of the general public. Since radiation workers derive benefit from their occupation, the standard for occupational exposure is less restrictive than for the general public. Persons who work with radiation are usually persons over 18 years of age, they are instructed in radiation protection practices, and they are provided with necessary surveying and monitoring instrumentation. Members of the general public, on the other hand, include the very sensitive young and have no regular supervision and monitoring of their radiation exposure. Consequently, levels for members of the general public are set lower than those for occupational exposure.

1. Recommendations

Basic recommendations of the NCRP (NCRP 39, 1971) fall into two categories, occupational and general public.

a) Occupational

Table 12.3 summarized the most recent NCRP occupational recommendations. Of particular note is the annual limit of 5 rem in any one year and the recommendations for partial body dose, all of which are greater than 5 rem in any one year.

It should be noted that the whole body limit is determined by the critical organs like the lens of the eyes, the blood-forming organs, and the gonads. Another interesting entry in the table is the recommendation for exposure to the fetus of not more than 0.5 rem in the gestation period. It is interesting to note that this current total-body occupational limit has been the recommendation since the 1950's. Figure 12.7 traces the downward trend in the recommended maximum permissible exposure levels since the early 1900's. The figure also summarizes the effect of radiation exposure at various dose levels.

b) General Public

The whole body limit for the general public or occasionally

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exposed individuals is 1/10 of the occupational limit, or 0.5 rem in any one year. Recognizing that students may require some radiation exposure in the course of their training and research, it is recommended that no more than 0.1 rem be received in addition to what they might receive as a member of the general public.

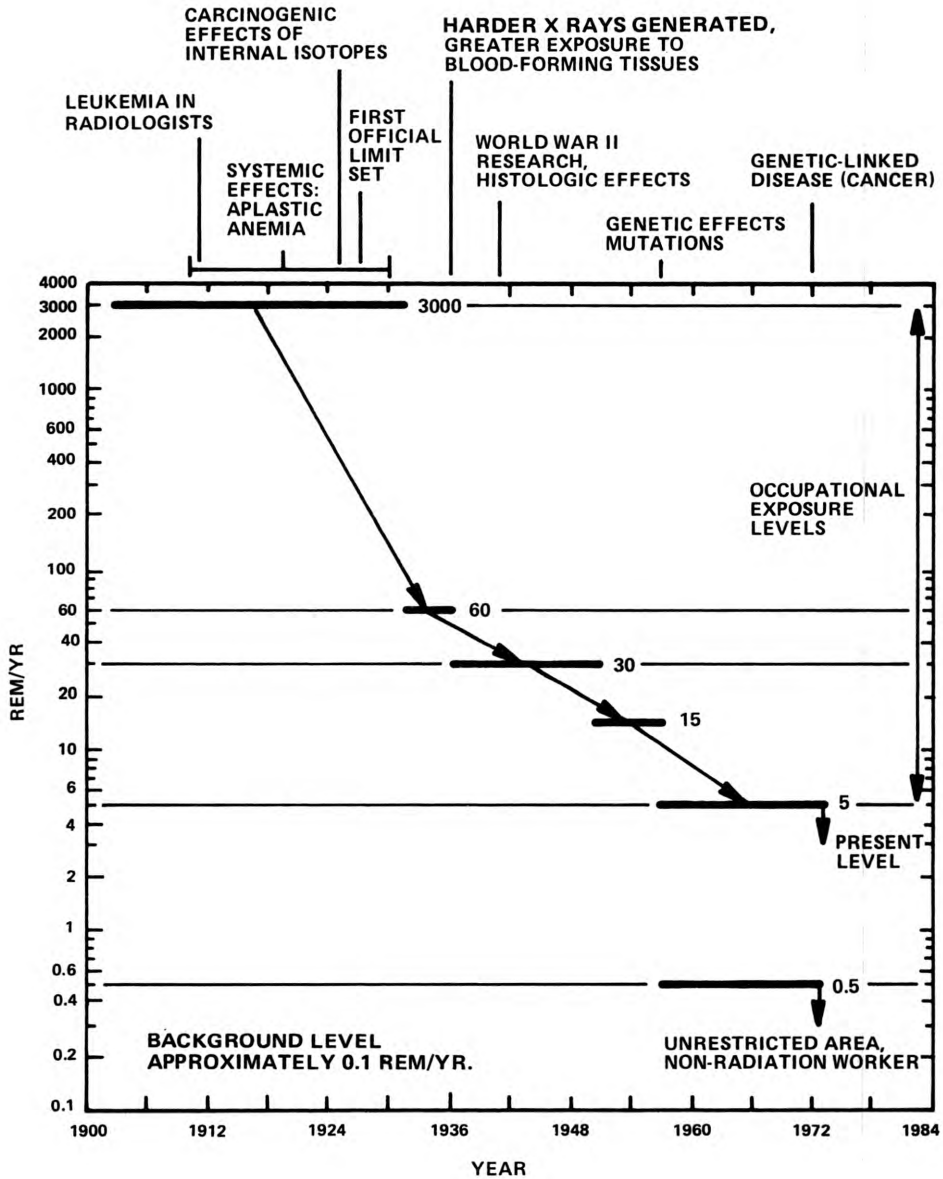


FIGURE 12.7 MAXIMUM PERMISSIBLE EXPOSURE LEVELS (LANZA, 1979).

When monitoring the radiation exposures of the general public, it frequently occurs that only pathways of radiation exposure are

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monitored, rather than individual exposures, and thus the value that is estimated for this type of measurement can only represent dose to a population average. Individuals will receive exposures both greater than and less than the average. It is felt that very few members of a group will receive exposures exceeding three times that of the group average. Therefore, the annual limit for an average range of population group has been reduced to one third, from 0.5 rem per year to 0.17 rem per year, to assure that the majority of the individuals will not exceed 0.3 rem per year.

It is interesting to note that NCRP recommendations for the general public apply only to total body exposure. In 1961, the Federal Radiation Council provided additional recommendations for the general public that included guidelines for thyroid, bone marrow, and bone as indicated in Table 12.4.

TABLE 12.4 FRC RADIATION PROTECTION GUIDES (RPG) FOR CERTAIN BODY ORGANS (1961).

ORGAN	RPG FOR INDIVIDUALS	RPG FOR AVERAGE OF A POPULATION GROUP
BONE MARROW	0.5 REM/YR	0.17 REM/YR
THYROID, BONE	1.5 REM/YR	0.5 REM/YR
BONE (ALTERNATE GUIDE)	0.003 g Ra-226*	0.001 g Ra-226*

* GUIDE REFERS TO QUANTITY OF Ra-226 IN THE ADULT SKELETON OR THE BIOLOGICAL EQUIVALENT OF THIS AMOUNT OF Ra-226.

2. Regulations

The radiation exposure limits specified by the NRC and agreement states are essentially the same as those of the NCRP, but differ in the following aspects:

a) NCRP states that the quarterly occupational dose may be as high as 3 rem per quarter. The regulatory agencies, on the other hand, prefer to state that the limit is 1.25 rem per quarter (5 rem/yr, 4 quarters/year) unless a record is kept of the maximum permissible accumulative dose and then the level might be as high as 3 rem.

b) In its Report 29, NCRP has specified 15 rem per year for the skin of the whole body, whereas the regulatory agencies use the older value of 30 rem per year.

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c) The regulatory agencies specify a value of 75 rem per year for the hands, forearms, feet and ankles. However, NCRP has indicated that since, in most cases, forearms will receive less than hands, and forearms contain more bone marrow, there is no reason for forearms to have an exposure larger than 30 rem per year and there is no reason for feet and ankles to be treated differently from other single organs.

d) NCRP specified 15 rem per year for the single organs while the regulatory agencies do not specify a limit for these organs.

e) NCRP has introduced limits for fertile women, the average radiation exposure to the unborn child in a population group, and students under 18 in the course of their training. These limits have not yet been incorporated into the regulation by the agencies.

3. Derived Limits - Internal Exposure

a) Maximum Permissible Body Burden (MPBB) - The dose rate from an internally hazardous radionuclide is proportional to the quantity or activity present in the organ of interest and the total dose is proportional to the cumulative activity in the organ of interest. Given a quality factor for the radiation, the dose equivalent rate and the total dose equivalent are proportional to the activity and cumulative activity respectively. If one assumes that a constant fraction of the total body content of the particular radionuclide is contained in this organ, then the dose equivalent rate is also proportional to the total quantity of radioactivity in the body. This quantity of radioactivity in the total body is called "Body Burden." Since dose and dose equivalent are proportional to body burden, then there should be a value of the total body burden that just delivers a maximum permissible dose equivalent. This quantity of radioactivity is referred to as the Maximum Permissible Body Burden (MPBB). Stated in another way, the MPBB is the quantity of radioactivity in the total body that will result in the maximum permissible dose rate to the critical organ. This critical organ is an organ that accumulates the radioactive material, is essential to the well being of the body, and is damaged by entry of the radioactivity. For bone-seeking radionuclides, such as Sr-90 and Pu-239, Occupational MPBB is based on a comparison with radium. Continuing studies have shown that a body burden of 0.1 microgram Ra-226 (0.1 micro Ci) is a reasonable limit that has not resulted in any observable effects in the individual. Thus, MPBB's for other bone seekers are set at the body content that corresponds to a bone burden that delivers the same dose equivalent rate as 0.1 microgram of Ra-226. Details on these concepts as well as other basic concepts, considerations and recommendations on radiation protection can be found in ICRP 26.

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b) Intake and Body Burden - Radioactive materials taken into the body by some route (ingestion, inhalation, or through the skin) are transported in the blood stream and then excreted or deposited in various organs and tissues. Radioactive materials in the body are lost by radioactive decay and by biological turnover; they are replaced by intake.

Once the maximum permissible body burden is determined for any particular radionuclide, it is then possible to calculate the concentrations of the isotope in air and water which would result in the accumulation in the body after continuous exposure of no more than the maximum permissible body burden. To do this, it is necessary to have information available on the body's metabolism of the material containing the radionuclide.

Detailed information on how these calculations are made can be found in a number of sources (e.g., Cember, 1976).

c) Maximum Permissible Concentration (MPC) - The intake of a radionuclide is proportional to the concentration in air, food or water. The Maximum Permissible Concentration (MPC) is a concentration which, if constantly maintained in air or water, will result in an intake to a standard or reference man producing the maximum permissible body burden in 50 years and therefore deliver the maximum permissible dose rate. Ordinarily, the maximum permissible body burden and maximum permissible concentration calculations are based on the so-called standard man or reference man who weighs 70 kg, breathes air at a rate of 20 cubic meters per day, and takes in 1.5 liters of water per day. One-half of the daily air and water are taken in during the working day and approximately one third of the weekly air and water are taken in during a 40-hour working week.

Maximum permissible concentrations have been computed by an ICRP committee and have just been published by ICRP and NCRP. In general, concentrations are calculated for an occupational 40-hour week for soluble and insoluble materials. Since in a 40-hour working week the reference individual consumes approximately 1/3 of the total air and water consumed during a full 168-hour week, ICRP and NCRP also present 168-hour occupational maximum permissible concentrations that are 1/3 (rounded off) those for a 40 hour week. Although ICRP and NCRP do not perform these calculations, it is suggested that an appropriate value for the general public would be 1/10th of the occupational 168-hour week.

The NRC and the various states have adopted the 40-hour occupational MPC for their regulations. NRC and state regulations also include non-occupational 168-hour week MPC's that are $1/30$ ($1/3 \times 1/10$) the occupational 40-hour concentration.

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d) Inert Gases - MPC Based on Dose from Immersion Rather Than Intake - Limits for mixtures are determined by considering for fractional part of each and setting the condition,

$$\sum_i \frac{\text{Conc}_i}{\text{MPC}_i} \leq 1 \quad (12.3)$$

For unknown mixtures, the following two rules are applied:

- i. Base MPC on the most hazardous nuclide, or,
- ii. If it is known that certain hazardous nuclides are absent, use the MPC for the most hazardous nuclide possibly present.

e) Derived Limits Neutron Fluxes - For neutrons it is frequently more convenient to measure neutron flux than it is to measure absorbed dose or dose equivalent. Given a neutron flux at a particular energy, it is possible to calculate the maximum dose equivalent rate in the body. Thus, for each neutron energy there is a maximum permissible flux density such that a maximum permissible dose equivalent is delivered under continued exposure.

f) Airborne Concentration Limit - Radon Daughters - A special limit has been devised in response to the special monitoring problems associated with the airborne radon daughters. The short-lived airborne radon daughters are usually measured by sampling and analyzing airborne particulate. Atmospheres may have various combinations of the members of the radon daughter decay series and each combination has a different dose per unit activity observed. Because a variety of possible mixtures of the various radioactive radon daughters might be found in the atmosphere, a special unit, the "Working Level" (WL), was devised to provide a meaningful expression of the concentration of radon daughter products in air without the necessity of specifying the proportion of each product. One WL is the concentration of radon daughter products in air that will result in the ultimate emission of 1.3×10^5 MeV of alpha energy. This is equivalent to the energy released by the decay products in equilibrium with 100 pCi of Rn-222 per liter of air. Cumulative exposure for occupational purposes is defined in terms of "Working Level Months" (WLM). Inhalation of air with radon daughters at a concentration of 1 WL for 170 work hours (1 working month) will result in a cumulative exposure of 1 WLM. OSHA standards require that occupational exposure to radon daughters be controlled so that no individual will receive an exposure of more than 2 WLM in any calendar quarter or 4 WLM in any calendar year. This corresponds to a maximum concentration of 0.33 WL for continuous 40 hour per week occupational exposure. (While NRC regulations specify a maximum permissible radon-222 concentration assuming that daughters products are in equilibrium, the regulations also recognize 0.33 WL as an equivalent standard.)

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g) Drinking Water - The Safe Drinking Water Act of 1974 directed the EPA to adopt regulations on various contaminants in drinking water, including radioactivity. Final regulations were published in July of 1976 and they became effective June 24, 1977.

12.2.2 Radiation Protection Practices

To create the safest conditions possible in areas where radiation or radioactivity is used, it is essential to keep radiation exposure as low as reasonably achievable. To do this, one must keep in mind at all times the type and the energy of the radiation being used.

Three practical guidelines for reducing one's exposure when working with radiation are time, distance and shielding.

Time: The total exposure is, of course, directly related to the time spent in the radiation field. Therefore, the work that must be accomplished should be done as efficiently as possible without sacrificing care.

Distance: The greater the distance from the radiation source, the smaller the exposure. For a point source, or one whose dimensions are small in comparison with its distance from a worker, the inverse square law applies, i.e., the intensity of a radiation field is inversely proportional to the square of the distance between the source and the observer or worker, or:

$$I(s) = \frac{I_0}{s^2} \quad (12.4)$$

where I_0 is the exposure rate in mR/hr for a given distance. For a point source of known activity, the exposure rate can be calculated for a given distance, s , in meters:

$$I_0 \text{ (mR/hr)} = \frac{n \cdot I_\gamma}{s^2} \quad (12.5)$$

where n is the source activity in mCi and I_γ (exposure rate constant) values can be found in many handbooks. Some useful values of I_γ are given in Table 12.5. The use of I_γ can be a help in doing rapid calculations of exposure. One must be careful that the units are accurately adhered to.

Shielding: The "effective" distance between a source and a worker can be increased by the use of an absorbing shield. The exact attenuation of the radiation depends on the nature of the radiation and on the composition and thickness of the shield.

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For gamma radiation, attenuation is exponential in nature and can be expressed by:

$$I(x) = I_0 e^{-\mu x} \quad (12.6)$$

where μ is the linear attenuation co-efficient and x is the distance traveled by the photon beam in the absorber. Values of μ can be found in many handbooks. (BRH, 1970).

TABLE 12.5 SPECIFIC EXPOSURE RATE CONSTANT (I_γ) IN UNITS OF R/(hr-Ci) AT ONE METER.

ISOTOPE	I_γ	ISOTOPE	I_γ
Co-57	0.097	I-125	0.0044
Co-60	1.30	I-131	0.21
Cr-51	0.018	Ra-226	0.825
Cs-137	0.323	Rb-86	0.05
Fe-59	0.64	Tc-99m	0.072
Ga-67	0.11	Xe-133	0.01

Some of the detailed responsibilities associated with the safe use of radiation include:

1. The design of buildings and equipment to facilitate safe handling of radiation and radioactivity.
2. The measurement of radiation levels during experiments, measurement of radiation exposure of personnel involved in such experiments, and the keeping of proper records of same.
3. Periodic surveys of all radioisotope laboratories to detect presence of loose radioactivity (contamination).
4. Proper disposal of radioactive waste.
5. Calibration of radiation measuring instruments.
6. Ability to cope with radiation emergencies, such as cuts contaminated by radioactivity and handling of persons exposed to radiation.
7. Education of all personnel concerning proper techniques for handling radioactivity and radiation-producing machines.
8. Design of better and safer procedures for all radiation work. (Wang, 1975).

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It is not possible to specify working rules that will apply in all situations. The level of radiation will determine the degree of precaution required. Table 12.6 is an example of a typical set of safety rules for a laboratory where radionuclides are used.

TABLE 12.6 SAFETY RULES FOR RADIONUCLIDE LABORATORY

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|---|--|
| <ol style="list-style-type: none">1. No smoking, eating, application of cosmetics, or storage of food is permitted in the area where radionuclides are used.2. Personal belongings (e.g., books and clothing) are not to be placed on laboratory work benches.3. No pipetting of radioactive material is to be done by MOUTH; a syringe or remote control equipment should be used.4. Trays, benches or hoods should be lined with absorbent paper before using radioactive material. This will confine spills and facilitate easy decontamination.5. Use hoods in a manner that their protective action is not compromised. Do not block air flow patterns. Do not cause eddies to be generated. Keep the opening to a minimum. Do not withdraw hands or objects from the hood quickly.6. Plastic or rubber gloves should be used whenever hand contamination is probable; gloves should be removed before leaving the area where radioactive material is being used.7. Hands should always be washed and monitored before leaving the laboratory.8. RADIOACTIVE WASTE:<ol style="list-style-type: none">A. All LIQUID radioactive waste material should be collected in a polyethylene container. This container should be labeled "CAUTION-RADIOACTIVE MATERIAL". | <p>When the container is full and ready to be picked up by Radiation Control for disposal, the label must show the TYPE of radioactive material, AMOUNT, DATE, and principal investigator's NAME.</p> <ol style="list-style-type: none">B. All DRY radioactive waste should be stored in properly covered waste containers which are lined with either a plastic bag or a heavy paper bag. This container should be labeled "CAUTION-RADIOACTIVE MATERIAL." The heavy paper bag is highly recommended where disposable pipets and glassware are used as this bag resists puncture much more than the plastic. When the bag is full and ready to be picked up by Radiation Control for disposal, the label must show the TYPE of radioactive material, AMOUNT, DATE and principal investigator's NAME. <ol style="list-style-type: none">9. Equipment and working areas and survey records should be monitored periodically for review by State and University Radiation Control personnel.10. All gamma emitters should be shielded so that the exposure rate is below 2mR/hr.11. All spills should be reported to the Radiation Control Office immediately. |
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The rules are set forth in an effort to minimize internal and external exposures to persons working in the laboratory and others, such as those who must handle radioactive waste.

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GLOSSARY

- A -

ABSORBED FRACTION: A term used in internal dosimetry. It is that fraction of the photon energy (emitted within a specified volume of material) which is absorbed by the volume. The absorbed fraction depends on the source distribution, the photon energy, and the size, shape, and composition of the volume.

ABSORPTION: The process by which radiation imparts some or all of its energy to any material through which it passes. (See COMPTON EFFECT, PHOTOELECTRIC EFFECT, and PAIR PRODUCTION.)

SELF ABSORPTION: Absorption of radiation emitted by radioactive nuclei, by the nuclei or atoms themselves; in particular, the absorption of radiation within a sample being assayed.

ABSORPTION COEFFICIENT: Fractional decrease in the intensity of a beam of x or gamma radiation per unit thickness (linear absorption coefficient), per unit mass (mass absorption coefficient), or per atom (atomic absorption coefficient) of absorber, due to deposition of energy in the absorber. The total absorption coefficient is the sum of the individual energy absorption processes (Compton effect, photoelectric effect, and pair production).

ATOMIC ABSORPTION COEFFICIENT: The linear absorption coefficient of a nuclide divided by the number of atoms per unit volume of the nuclide. It is equivalent to the nuclide's total cross section for the given radiation.

COMPTON ABSORPTION COEFFICIENT: That fractional decrease in the energy of a beam of x and gamma radiation due to the deposition of the energy to electrons produced by Compton effect in an absorber. (See SCATTERING, COMPTON.)

LINEAR ABSORPTION COEFFICIENT: A factor expressing the fraction of a beam of x or gamma radiation absorbed in unit thickness of material. In the expression $I = I_0 e^{-\mu x}$, I_0 is the initial intensity, I the intensity of the beam after passage through thickness of the material x , and μ is the linear absorption coefficient.

MASS ABSORPTION COEFFICIENT: The linear absorption coefficient per centimeter divided by the density of the absorber in grams per cubic centimeter. It is frequently expressed as μ/ρ , where μ is the linear absorption coefficient and ρ the absorber density.

ABSORPTION RATIO, DIFFERENTIAL: Ratio of concentration of a nuclide

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in a given organ or tissue to the concentration that would be obtained if the same administered quantity of this nuclide were uniformly distributed throughout the body.

ACCELERATOR (PARTICLE ACCELERATOR): A device for imparting large kinetic energy to electrically charged particles such as electrons, protons, deuterons, and helium ions. Common types of particle accelerators are direct voltage accelerators (including Van de Graaff, Cockcroft-Walton, Dynamitron, resonant transformer and insulating core transformer), cyclotrons (including synchrocyclotrons and isochronous cyclotrons), betatrons, and linear accelerators, (individual accelerators listed alphabetically throughout glossary).

ACTIVATION: The process of inducing radioactivity by irradiation.

ACTIVITY: The number of nuclear transformations occurring in a given quantity of material per unit time (see CURIE).

ALPHA PARTICLE: A charged particle emitted from the nucleus of an atom having a mass and charge equal in magnitude of a helium nucleus; i.e., two protons and two neutrons.

ALVEOLI: The terminal air sacs of the lungs.

ALUMINUM EQUIVALENT: The thickness of aluminum affording the same attenuation, under specified conditions, as the material in question.

AMPERE: The unit of current that, when flowing through each of two long parallel wires separated by one meter in free space, results in a force between the two wires (due to their magnetic fields) of 2×10^{-7} newtons for each meter of length.

AMPLIFICATION: As related to radiation detection instruments, the process (gas, electronic, or both) by which ionization effects are magnified to a degree suitable for their measurement.

AMPLIFIER, LINEAR: A pulse amplifier in which the output pulse height is proportional to an input pulse height for a given pulse shape up to a point at which the amplifier overloads.

AMPLIFIER, PULSE: An amplifier, designed specifically to amplify the intermittent signals of a nuclear detector, incorporating appropriate pulse-shaping characteristics.

ANALYSIS, ACTIVATION: A method of chemical analysis, especially for small traces of material, based on the detection of characteristic radiations following a nuclear bombardment.

ANALYSIS, FEATHER: A technique for the determination of the range in aluminum of the beta particles of a radionuclide by comparison of the absorption curve of a reference source, usually ^{210}Bi .

ANALYSIS, ISOTOPE DILUTION: A method of chemical analysis in which a known amount of labelled compound having a known specific activity is added to a sample. Portions of the mixture are then isolated and the

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specific activity of that sample measured.

ANALYZER, PULSE HEIGHT: An electronic circuit which sorts and records the pulses according to height.

ANEMIA: Deficiency of blood as a whole, or deficiency in the number of the red corpuscles or of the hemoglobin.

ANGSTROM UNIT: One angstrom unit equal 10^{-8} cm.

ANION: Negatively charged ion.

ANNIHILATION (ELECTRON): An interaction between a positive and a negative electron in which they both disappear; their energy, including rest energy, being converted into electromagnetic radiation (called annihilation radiation).

ANODE: Positive electrode; electrode to which negative ions are attracted.

ANTIMATTER (ANTIPARTICLES): Matter in which the ordinary nuclear particles (neutrons, protons, electrons, etc.) are conceived of as being replaced by their corresponding antiparticles (antineutrons, antiprotons, positrons, etc.). An antihydrogen atom, for example, would consist of a negatively charged antiproton with an orbital positron. Normal matter and antimatter would mutually annihilate each other upon contact, being converted totally into energy.

ATOM: A unit of matter consisting of a single nucleus surrounded by a number of electrons equal to the number of protons in the nucleus.

ATOMIC MASS: The mass of a neutral atom of a nuclide, usually expressed in terms of "atomic mass units." The "atomic mass unit" is one-twelfth the mass of one neutral atom of carbon-12; equivalent to 1.6604×10^{-24} gm.

ATOMIC NUMBER: The number of protons in the nucleus of a neutral atom of a nuclide.

ATOMIC WEIGHT: The weighted mean of the masses of the neutral atoms of an element expressed in atomic mass units.

ATTENUATION: The process by which a beam of radiation is reduced in intensity when passing through some material. It is the combination of absorption and scattering processes through matter.

ATTENUATION COEFFICIENT, COMPTON: The fractional number of photons removed from a beam of radiation per unit thickness of a material as a result of Compton effect interactions.

ATTENUATION COEFFICIENT, LINEAR: The fractional number of photons removed from a beam of radiation per unit thickness of a material due to all absorption and scattering processes.

ATTENUATION COEFFICIENT, PAIR PRODUCTION: That fractional decrease in the intensity of a beam of ionizing radiation due to photoelectric effect.

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ATTENUATION FACTOR: A measure of the opacity of a layer of material for radiation traversing it; the ratio of the incident intensity to the transmitted intensity. It is equal to I_0/I , where I_0 and I are the intensities of the incident and emergent radiation, respectively. In the usual sense of exponential absorption ($I = I_0 e^{-\mu x}$), the attenuation factor is $e^{-\mu x}$, where x is the thickness of the material and μ is the absorption coefficient.

AUGER EFFECT: The emission of an electron from the extra-nuclear portion of an excited atom when the atom undergoes a transition to a less excited state.

AUTOFLUOROSCOPE: A device of visualizing the spatial distribution of a radionuclide within an organ or gland in the body. The autofluoroscope uses a multielement stationary detector composed of individual NaI(tl) crystals. An image of the radionuclide distribution is obtained on a retention oscilloscope, or other readout devices.

AUTORADIOGRAPH: Record of radiation from radioactive material in an object, made by placing the object in close proximity to a photographic emulsion.

AVALANCHE: The multiplicative process in which a single charged particle accelerated by a strong electric field produces additional charged particles through collision with neutral gas molecules. This cumulative increase of ions is also known as "Townsend ionization" or "Townsend avalanche."

AVERAGE LIFE (MEAN LIFE): The average of the individual lives of all the atoms of a particular radioactive substance.

AVOGADRO'S NUMBER (AVOGADRO CONSTANT): Number of atoms in a gram atomic weight of any element; also the number of molecules in a gram molecular weight of any substance. It is numerically equal to 6.023×10^{23} on the unified mass scale. (Symbol: N_A).

- B -

BACKGROUND COUNTING RATE: That radiation component detected by instrument due to cosmic rays, to radioactive materials in the vicinity, and to a slight radioactive contamination of the materials of which the instruments are made.

BACKSCATTER: The deflection of radiation by scattering processes through angles greater than 90 degrees, with respect to the original direction of motion in the laboratory frame of references.

BARN: Unit expressing the probability of a specific nuclear reaction, in terms of cross-sectional area. Numerically, it is 10^{-24} cm².

BARRIERS, PROTECTIVE: Barriers of radiation-absorbing material,

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such as lead, concrete, and plaster, used to reduce radiation exposure.

BARYON: One of a class of heavy elementary particles which includes neutrons, protons, and hyperons.

BEAM: A unidirectional or approximately unidirectional flow of electromagnetic radiation or of particles.

BEAM HOLE (GLORY HOLE): Hole through the shield, and usually through the reflector, of a reactor to permit the escape of a beam of radiation, in particular a beam of fast neutrons, for experimental purposes.

BETA DECAY: Radioactive decay in which a beta particle is emitted or in which orbital electron capture occurs.

BETA PARTICLE: An electron, of either positive or negative charge, which has been emitted by an atomic nucleus or neutron in a nuclear transformation.

BETATRON: A magnetic induction acceleration which makes use of a varying magnetic field to accelerate electrons. Electrons are injected into a toroidal vacuum chamber which is between the poles of an iron-core magnet. The rate of change of the magnet flux and magnetic field at the orbit radius are related to maintain a constant radius for the accelerating electrons.

BF₃ COUNTER: A proportional counter designed to count neutrons. The counter is filled with the gas boron trifluoride. A neutron hitting boron nucleus will often result in the emission of a 2 MeV energetic alpha particle which ionizes heavily in the counter producing a large pulse which can be distinguished electronically from other smaller pulses such as those produced by beta rays. Often the gas is enriched in the isotope boron-10 to increase the sensitivity of the counter since this isotope is the principal contributor to the reaction.

BIOLOGIC EFFECTIVENESS OF RADIATION: (See RELATIVE BIOLOGICAL EFFECTIVENESS.) It is 1.443 times the radioactive half-life.

BLOOD DYSCRASIA: Any persistent change from normal of one or more of the blood components.

BODY BURDEN: The total quantity of radionuclide present in the body.

BODY BURDEN, MAXIMUM PERMISSIBLE: That body burden of a radionuclide which, if maintained at a constant level would produce the maximum permissible dose equivalent in the critical organ.

BONE SEEKER: Any compound or ion which migrates in the body preferentially into bone.

BRACHYTHERAPY: Therapy at short distances with beta or gamma radiation. Implantation or placement therapy with needles, inserts, or other such applications containing radioactive materials. Useful in the treatment of various diseases.

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BRAGG-GRAY RELATION: A relationship stating that the ionization produced in small gas-filled cavity in a homogeneous medium by a uniform field of ionizing radiation is proportional to the absorbed dose in that medium.

BRANCHING: The occurrence of two or more modes by which a radionuclide can undergo radioactive decay. For example, RaC can undergo α^+ or α^- decay, Cu can undergo α^+ , α^- , or electron capture decay. The fraction disintegrating by a particular mode is the "branching fraction" for that mode. The "branching ratio" is the ratio of two specified branching fractions (also called multiple disintegration).

BREMSSTRAHLUNG: Secondary photon radiation produced by deceleration of charged particles passing through matter.

BRITISH THERMAL UNIT (BTU): The quantity of heat required to increase the temperature of one pound of water one degree Fahrenheit at atmospheric pressure; approximately 252 gram-calories.

BUILD-UP FACTOR: The ratio of the intensity of x or gamma radiation (both primary and scattered) at a point in an absorbing medium to the intensity of only the primary radiation. This factor has particular application for "broad beam" attenuation. "Intensity" may refer to energy flux, dose, or energy absorption.

BURIAL GROUND (GRAVEYARD): A place for burying unwanted radioactive objects to prevent escape of their radiations, the earth or water acting as a shield. Such objects must be placed in watertight, noncorrodible containers so the radioactive material cannot leak out and invade underground water supplies.

BURNUP: Induced nuclear transformation of atoms during reactor operation. The term may be applied to fuel or other materials. (See also BURNUP, SPECIFIC).

-C-

CADMIUM CUTOFF, EFFECTIVE: In a given experimental configuration: the energy value determined by the condition that detector response would be unchanged if the cadmium cover surrounding the detector were replaced by a fictitious cover opaque to neutrons with energy below this value and transparent to neutrons with energy above this value.

CADMIUM RATIO: The ratio of the response of a bare neutron detector to its response under the same conditions but when covered with cadmium of a specified thickness.

CAPTURE, ELECTRON: A mode of radioactive decay involving the capture of an orbital electron by its nucleus. Capture from a particular electron shell is designated as "K-electron capture," "L-electron capture," etc.

CAPTURE, K-ELECTRON: Electron capture from the K-shell by the nucleus of the atom. Also loosely used to designate any orbital electron capture process.

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CAPTURE, RADIATIVE: The process by which a nucleus captures an incident particle and loses its excitation energy immediately by the emission of gamma radiation.

CAPTURE, RESONANCE: An inelastic nuclear collision occurring when the nucleus exhibits a strong tendency to capture incident particles or photons of particular energies.

CARCINOGENIC: Capable of producing cancer.

CARRIER: A quantity of non-radioactive or non-labeled material of the same chemical composition as its corresponding radioactive or labeled counterpart. When mixed with the corresponding labeled material, so as to form a chemically inseparable mixture, the carrier permits chemical (and some physical) manipulation of the mixture with less label or radioactivity loss than would be true for the undiluted label or radioactivity.

CARRIER, HOLD-BACK: The inactive isotope or isotopes of a radioactive element, or an element of similar properties, or some reagent which may be used to diminish the amount of the radionuclide coprecipitated or absorbed in a chemical reaction.

CARRIER, FREE: An adjective applied to one or more radioactive isotopes of an element in minute quantity, essentially undiluted with stable isotope carrier.

CATALYST: A substance which alters the velocity of a chemical reaction (positive catalysts increase the velocity) yet may be recovered practically unchanged after the reaction has occurred.

CATARACT: A clouding of the crystalline lens of the eye which obstructs the passage of light.

CATHODE: Negative electrode; electrode to which positive ions are attracted.

CATION: Positively charged ion.

CELL (REACTOR): One of a set of elementary regions in a heterogeneous reactor, each of which has the same geometrical form, material composition, and neutron characteristics.

CELL, HOT: A heavily shielded enclosure for highly radioactive materials. It may be used for their handling or processing by remote means or for their storage. (Also called cell, shielded).

CHAMBER, CLOUD: A device for observing the paths of ionizing particles. It is based on the principle that supersaturated vapor condenses more readily on ions than on neutral molecules.

CHAMBER, IONIZATION: An instrument designed to measure a quantity of ionizing radiation in terms of the charge of electricity associated with ions produced within a defined volume.

AIR-WALL IONIZATION CHAMBER: Ionization chamber in which the

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materials of the wall and electrodes are so selected as to produce ionization essentially equivalent to that in a free-air ionization chamber. This is possible only over limited ranges of photon energies. Such a chamber is more appropriately termed an "air-equivalent" ionization chamber.

EXTRAPOLATION IONIZATION CHAMBER: An ionization chamber with electrodes whose spacing can be adjusted and accurately determined to permit extrapolation of its reading to zero chamber volume.

FREE-AIR IONIZATION CHAMBER: An ionization chamber in which a delimited beam of radiation passes between the electrodes without striking them or other internal parts of the equipment. The electric field is maintained perpendicular to the electrodes in the collecting region. As a result, the ionized volume can be accurately determined from the dimensions of the collecting electrode and the limiting diaphragm. This is the basic standard instrument for x-ray dosimetry within the range of 5 to 1400 kVp.

THIMBLE IONIZATION CHAMBER: A small cylindrical or spherical ionization chamber, usually with walls of organic material.

TISSUE-EQUIVALENT IONIZATION CHAMBER: An ionization chamber in which the material of the walls, electrodes, and gas are so selected as to produce ionization essentially equivalent to that characteristic of the tissue under consideration. In some cases it is sufficient to have only tissue equivalent walls, and the gas may be air, provided the air volume is negligible. The essential point in this case is that the contribution to the ionization in the air made by ionizing particles originating in the air is negligible, compared to that produced by ionizing particles characteristic of the wall material.

CHAMBER, POCKET: A small, pocket-sized ionization chamber used for monitoring radiation exposure of personnel. Before use, it is given a charge and the amount of discharge is a measure of the radiation exposure.

CHARGE: The fissionable material or fuel placed in a reactor to produce a chain reaction. To assemble the charge in a reactor.

CHARGE, SPACE: The electric charge carrier by a cloud or stream of electrons or ions in a vacuum or a region of low gas pressure, when the charge is sufficient to produce local changes in the potential distribution. It is of importance in thermionic tubes, photoelectric cells, ion accelerators, etc.

CHEMICAL DOSIMETER: A self-indicating device for determining total (or accumulated) radiation exposure dose based on color changes accompanying chemical reactions induced by the radiation.

CHEMICAL (ISOTOPIC) EXCHANGE: A process in which atoms (isotopes) of the same element in two different molecules exchange places.

CERENKOV RADIATION: Blue light emitted when a charge particle moves

GLOSSARY

in a transparent medium with a speed greater than that of light in the same medium.

CIRCUIT, ANTICOINCIDENCE: A circuit with two input terminals which delivers an output pulse if one input terminal receives a pulse, but delivers no output if pulses are received by both input terminals simultaneously or within an assignable time interval.

CIRCUIT, COINCIDENCE: An electronic circuit that produces a usable output pulse only when each of two or more input circuits receives pulses simultaneously or within an assignable time interval.

CIRCUIT, INTEGRATING: An electronic circuit which records the total number of ions or events collected for a given time from which an average value for the number of ions or events per unit time can be found.

CIRCULAR MIL: An area equal to the area contained in a circle of one mil in diameter or 7.854×10^{-7} of a square inch.

CLADDING (CLAD): An external layer of material applied directly to nuclear fuel or other material to provide protection from a chemically reactive environment, to provide containment of radioactive products produced during the irradiation of the composite, or to provide structural support.

COCKCROFT-WALTON ACCELERATOR: A device for accelerating charged particles by application of a very high direct-current voltage to a stream of ions in a straight insulated tube. The high voltage is obtained through a number of rectifiers and capacitors arranged in a series-coupled-voltage multiplier circuit.

COINCIDENCE: The occurrence of counts in two or more detectors simultaneously or within an assignable time interval. A true coincidence is one that is due to the incidence of a single particle or of several genetically related particles. An accidental, chance or random coincidence is one that is due to the accidental occurrence of unrelated counts in the separate detectors. An anticoincidence is the occurrence of a count in a specified detector unaccompanied simultaneously or within an assignable time interval by a count in other specified detectors. A delayed coincidence is the occurrence of a count in one detector at a short, but measurable, time after a count in another detector. The two counts are due to a genetically related occurrence, such as successive events in the same nucleus.

COLLIMATOR: A device for confining the elements of a beam within an assigned solid angle.

COLLISION: Encounter between two particles (including photons) which changes the existing momentum and energy conditions. The products of the collision need not be the same as the initial systems.

ELASTIC COLLISION: A collision in which there is not change either in the internal energy of each participating system or in the sum of each participating system or in the sum of their kinetic energies of translation.

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INELASTIC COLLISION: A collision in which there are changes both in the internal energy of one or more of the colliding systems and in the sums of the kinetic energies of translation before and after the collision.

COLUMN, THERMAL: A column or large body of moderator, such as graphite, extending away from the active section of a nuclear reactor to provide near its other end (for experimental purposes) a flux of thermal neutrons of high cadmium ratio; i.e., containing few virgin and epithermal neutrons.

COMPTON EFFECT: An attenuation process observed for x or gamma radiation in which an incident photon interacts with an orbital electron of an atom to produce a recoil electron and a scattered photon of energy less than the incident photon.

CONDENSER R-METER: An instrument consisting of an "air-wall" ionization chamber together with auxiliary equipment for charging and measuring its voltage. It is used as an integrating instrument for measuring the exposure of x or gamma radiation in roentgens, (R). (See CHAMBER, IONIZATION.)

CONTACT HAZARD, RADIATION or CONTAMINATION: Radiation hazard resulting from direct bodily contact with containments, or from prolonged contact with a sealed source of gamma radiation.

CONTAMINATION, RADIOACTIVE: Deposition of radioactive material in any place where it is not desired, particularly where its presence may be harmful. The harm may be in vitiating an experiment or a procedure, or in actually being a source of danger to personnel.

CONTROL: The purposeful variation of the reactivity of a reactor. "Absorber control" is obtained by varying the amount of the neutron absorbers within the reactor. "Configuration control" is obtained by changing the geometry of the reactor.

CONTROL SYSTEM: A coordinated group of components designed to exert a directing influence on other components. A system of apparatus for controlling the rate of reaction in a nuclear reactor. The term may refer to all apparatus provided for this purpose or to one of several essentially independent arrangements, such as a regulating system and safety system. A reaction may be controlled automatically by a servo system that adjusts the control elements to maintain the flux level near a desired value. A reactor may have a tendency toward stability because of self-regulation, but this quality of stability ordinarily is not considered part of the control system.

CONTROLLED AREA: A defined area in which the occupational exposure of personnel (to radiation) is under the supervision of the Radiation Protection Supervisor.

CONVERSION (REACTOR TECHNOLOGY): Nuclear transformation of a fertile substance into a fissile substance.

CONVERSION, INTERNAL: A mode of radioactive decay in which the gamma rays from excited nuclei cause the ejection of orbital electrons from the atom. The ratio of the number of internal conversion electrons

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to the number of gamma quanta emitted in the de-excitation of the nucleus is called the "conversion ratio."

CONVERSION RATIO: The ratio of the number of fissile nuclei produced by conversion to the number of fissile nuclei destroyed. The term can refer to an instant of time or to a period of time.

CONVERTER REACTOR: The difference between "converter" and "breeder" reactor is that a converter produces fissile atoms from fertile atoms, but has a conversion ratio less than one. A breeder reactor has a conversion ratio greater than one and therefore produces more fissile atoms than it consumes.

COOLANT: A substance, usually liquid or gas, used for cooling any part of a reactor in which heat is generated. Such parts include not only the core but also the reflector, shield, and other elements that may be heated by absorption of radiation.

CORE: In a nuclear reactor, the region containing the fissionable material. The body of fuel or moderator and fuel in a nuclear reactor. It does not include the fuel outside the active section in a circulating reactor. Identical with active lattices in a reactor. In a heterogeneous reactor, the region containing fuel-bearing cells.

COSMIC RAY: High-energy particulate and electromagnetic radiations which originate outside the earth's atmosphere.

COULOMB: Unit of electrical charge in the MKSA system of units. A quantity of charge equal to one ampere second.

COUNT (Radiation Measurements): The external indication of a device designed to enumerate ionizing events. It may refer to a single detected event or to the total number registered in a given period of time. The term often is erroneously used to designate a disintegration, ionizing event, or voltage pulse.

COUNTER, GAS FLOW: A device in which an appropriate atmosphere is maintained in the counter tube by allowing a suitable gas to flow slowly through the sensitive volume.

COUNTER, GEIGER-MUELLER: Highly sensitive, gas-filled radiation-measuring device. It operates at voltages sufficiently high to produce avalanche ionization.

COUNTER, PROPORTIONAL: Gas-filled radiation detection device; the pulse produced in proportion to the number of ions formed in the gas by the primary ionizing particle.

COUNTER, SCINTILLATION: The combination of phosphor, photomultiplier tube, and associated circuits for counting light emissions produced in the phosphors.

COUNTING, COINCIDENCE: A technique in which particular types of events are distinguished from background events by coincidence circuits which register coincidences caused by the type of events under consideration.

GLOSSARY

COUNTING RATE: The average rate, as a time function, at which ionization events are indicated by an electronic circuit.

COUNTING RATEMETER: An instrument which gives a continuous indication of the average rate of ionization events.

CRITICAL: Capable of sustaining (at constant level) a chain reaction. "Prompt critical" means sustaining a chain reaction without the aid of delayed neutrons.

CRITICAL SIZE: Any one of a set of physical dimensions of the core and reflector of a nuclear reactor maintaining a critical chain reaction, the material and structure of the core and the reflector having been specified.

CROSS SECTION, CAPTURE: The probability that a nucleus will capture an incident particle. The unit of cross section is commonly the barn (10^{-24} cm²).

CROSS SECTION, NUCLEAR: The probability that a certain reaction between a nucleus and an incident particle or photon will occur. It is expressed as the effective "area" the nucleus presents for the reaction. "Macroscopic cross section" refers to the cross section per unit volume (preferably) or per unit mass. "Microscopic cross section" is the cross section of one atom or molecule. (See BARN and CROSS SECTION CAPTURE.)

CURIE: The special unit of activity. One curie equals 3.700×10^{10} nuclear transformations per second. (Abbreviated Ci). Several fractions of the curie are in common usage.

CYCLOTRON: A particle accelerator which uses a magnetic field to confine a positive ion beam to a plane while an alternating electric field accelerates the ions in a spiral path. An RF voltage is applied between one or two hollow semicircular electrodes called "dees" at the frequency at which the ions rotate (which is constant in the conventional cyclotron.) As the voltage between the dees alternates, particles are accelerated as they enter and leave the dees.

-D-

DAUGHTER: Synonym for decay product.

DECAY, RADIOACTIVE: Disintegration of the nucleus of an unstable nuclide by spontaneous emission of charged particles and/or photons.

DECAY, CONSTANT: The fraction of the number of atoms of a radioactive nuclide which decay in unit time. Symbol: λ . (See DISINTEGRATION CONSTANT.)

DECAY CURVE: A curve showing the relative amount of radioactive substance remaining after any time interval. (See DISINTEGRATION CONSTANT.)

GLOSSARY

DECAY PRODUCT: A nuclide resulting from the radioactive disintegration of a radionuclide, formed either directly or as the result of successive transformations in a radioactive series. A decay product may be either radioactive or stable.

DECONTAMINATION: The reduction or removal of contaminating radioactive material from a structure, area, object, or person. Decontamination may be accomplished by (1) treating the surface so as to remove or decrease the contamination; (2) letting the material stand so that the radioactivity is decreased as a result of natural decay; and (3) covering the contamination so as to attenuate the radiation emitted. Radioactive material removed in process (1) must be disposed of by burial on land or at sea or in some other suitable way. **DECONTAMINATION FACTOR:** The ratio of the amount of undesired radioactive material initially present to the amount remaining after a suitable processing step has been completed. Decontamination factors may refer to the reduction of some particular type of radiation, or to the gross measurable radioactivity.

DELAYED NEUTRON: Neutrons emitted by excited nuclei formed in a radioactive process; so-called because they are emitted an appreciable time after fission. They are important in the control of nuclear reactors.

DELTA RAY: Any secondary ionizing particle ejected by recoil when a primary ionizing particle passes through matter.

DENSITOMETER: Instrument utilizing a photocell to determine the degree of darkening of developed photographic film.

DENSITY (PHOTOGRAPHIC): Used to denote the degree of darkening of photographic film. $D = \log(I_0/I)$ where I_0 is the incident intensity and I the transmitted intensity.

DEPLETION: Reduction of the concentration of one or more specified isotopes in a material or in one of its constituents.

DETECTOR, 1/V: A neutron detector for which the cross section of the detection reaction varies inversely with neutron speed.

DETECTOR, RADIATION: Any device for converting radiant energy to a form more suitable for observation. An instrument used to determine the presence, and sometime the amount, of radiation.

DEUTERIUM: A heavy isotope of hydrogen with one proton and one neutron in the nucleus. (Symbol: ^2H or D).

DEUTERON: Nucleus of a deuterium atom. Direct voltage accelerator (potential drop accelerator): An accelerator which uses a constant voltage to accelerate particles and is typically constructed with an ion or electron source inside a "terminal," which operates at a very high voltage with respect to the target area, which is at ground potential. Usually named according to the type of power supply used.

DISINTEGRATION, NUCLEAR: A spontaneous nuclear transformation (radioactivity) characterized by the emission of energy and/or mass from the nucleus. When numbers of nuclei are involved, the process

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is characterized by a definite half-time.

DOPPLER BROADENING: In spectroscopy, the observed broadening of a spectral line resulting from the thermal motion of the molecules, atoms, or nuclei. In reactor technology, it is the observed broadening of the energy width of a cross section resonance resulting from the thermal motion of the target particles.

DOPPLER EFFECT: The change in the observed wavelength of a radiation which results from the motion of its source relative to the observer.

DOSE: A general term denoting the quantity of radiation or energy absorbed. For special purposes it must be appropriately qualified. If unqualified, it refers to absorbed dose.

ABSORBED DOSE: The energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest. The unit of absorbed dose is the rad. One rad equals 100 ergs per gram. (see RAD.)

CUMULATIVE DOSE (RADIATION): The total dose resulting from repeated exposures to radiation.

DEPTH DOSE: The radiation dose delivered at a particular depth beneath the surface of the body. It is usually expressed as a percentage of surface dose.

DOSE EQUIVALENT (DE): A quantity used in radiation protection. It expresses all radiation on a common scale for calculating the effective absorbed dose. It is defined as the product of the absorbed dose in rads and certain modifying factors. (The unit of dose equivalent is the rem.)

EXIT DOSE: Dose of radiation at surface of body opposite to that on which the beam is incident.

INTEGRAL DOSE (VOLUME DOSE): A measure of the total energy absorbed by a subject or object during exposure to radiation. (see GRAM-RAD.)

MAXIMUM PERMISSIBLE DOSE EQUIVALENT (MPD): The greatest dose equivalent that a person or specified part thereof shall be allowed to receive in a given period of time.

MEDIAN LETHAL DOSE (MLD): Dose of radiation required to kill, within a specified period, 50 percent of the individuals in a large group of animals or organisms. Also called the LD 50.

PERMISSIBLE DOSE: The dose of radiation which may be received by an individual within specified period with expectation of no significantly harmful result.

SKIN DOSE (RADIOLOGY): Absorbed dose at center of irradiation field on skin. It is the sum of the dose in air and scatter from body parts.

GLOSSARY

THRESHOLD DOSE: The minimum absorbed dose that will produce a detectable degree of any given effect.

TISSUE DOSE: Absorbed dose received by tissue in the region of interest, exposed in rads. (See DOSE and RAD.)

DOSE, FRACTIONATION: A method of administering radiation in which relatively small doses are given daily or at longer intervals.

DOSE PROTRACTION: A method of administering radiation by delivering it continuously over a relatively long period at a low dose rate.

DOSE METER, INTERGRATING: Ionization chamber and measuring system designated for determining total radiation administered during an exposure. In medical radiology the chamber is usually designed to be placed on the patient's skin. A device may be included to terminate the exposure when it has reached a desired value.

DOSE RATE: Absorbed dose delivered per unit time.

DOSE RATEMETER: Any instrument which measures radiation dose rate.

DOSIMETER: Instrument to detect and measure accumulated radiation exposure. In common usage, a pencil-size ionization chamber with a self-reading electrometer, used for personnel monitoring.

DOSIMETRY: The theory and application of the principles and techniques involved in the measurement and recording of radiation doses. Its practical aspect is concerned with the use of various types of radiation instruments with which measurements are made.

DOSIMETRY, PHOTOGRAPHIC: Determination of cumulative radiation dose with photographic film and density measurement.

DYNAMITRON: A particle accelerator using a voltage multiplying circuit with the stage driven by high voltage capacitors in parallel. A radio frequency power source is used to charge the capacitors.

DYNE: The unit of force which, when acting upon a mass of one gram, will produce an acceleration of one centimeter per second.

-E-

EFFICIENCY (COUNTERS): A measure of the probability that a count will be recorded when radiation is incident on a detector. Usage varies considerably, so it is well to ascertain which factors (window transmission, sensitive volume, energy dependence, etc.) are included in a given case.

ELECTRODE: A conductor used to establish electrical contact with a nonmetallic part of a circuit.

ELECTROMAGNETIC RADIATION: Radiation consisting of alternating electric and magnetic fields at right angles to each other which propagate energy through space in small bundles known as photons or

GLOSSARY

quanta. Gamma and X radiation are examples of electromagnetic radiation.

ELECTROMETER: Electrostatic instrument for measuring the difference in potential between two points. Used to measure change of electric potential of charged electrodes resulting from ionization produced by radiation.

ELECTROMOTIVE FORCE: Potential difference across electrodes tending to produce an electric current.

ELECTRON: A stable elementary particle having an electric charge equal to $+1.60210 \times 10^{-19}$ C. and a rest mass equal to 9.1091×10^{-31} KG.

ELECTRON CAPTURE: A mode of radioactive decay involving the capture of an orbital electron by its nucleus. Capture from a particular electron shell is designated as K-electron capture, L-electron capture, etc.

ELECTRON VOLT: A unit of energy equivalent to the energy gained by an electron in passing through a potential difference of one volt. Larger multiple units of the electron volt are frequently used: keV for thousand or kilo electron volts; MeV for million or mega electron volts. (Abbreviated: eV, $1 \text{ eV} = 1.6 \times 10^{-19} \text{ J.}$)

ELECTROSCOPE: Instrument for detecting the presence of electric charges by the deflection of charged bodies.

ELECTROSTATIC FIELD: The region surrounding an electric charge in which another electric charge experiences a force.

EMULSION, NUCLEAR: A photographic emulsion specially designed to permit observation of the individual tracks of ionizing particles.

END PRODUCT: The stable nuclide that is the final member of a radioactive series.

ENERGY: Capacity for doing work. "Potential energy" is the energy inherent in a mass because of its spatial relation to other masses. "Kinetic energy" is the energy possessed by a mass because of its motion; MKSA units: $\text{kg-m}^2/\text{sec}^2$ or joules.

BINDING ENERGY: The energy represented by the difference in mass between the sum of the component parts and the actual mass of the nucleus.

EXCITATION ENERGY: The energy required to change a system from its ground state to an excited state. Each different excited state has a different excitation energy.

IONIZING ENERGY: The average energy lost by ionizing radiation in producing an ion pair in a gas. For air, it is about 33.73 eV.

RADIANT ENERGY: The energy of electromagnetic radiation, such as radio waves, visible light, X and gamma rays.

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REACTION ENERGY (NUCLEAR): In the disintegration of a nucleus, it is equal to the sum of the kinetic or radiant energies of the products. If any product of a specified reaction is in an excited nuclear state, the energy of subsequently emitted gamma radiation is not included in the sum. The "ground-state nuclear radiation energy" is the reaction energy when all reactant and product nuclei are in their ground states. (Symbol: Q).

ENERGY DEPENDENCE: The characteristic response of a radiation detector to a given range of radiation energies or wave lengths compared with the response of a standard free-air chamber.

ENERGY FLUENCE: The sum of the energies, exclusive of rest energies, of all particles passing through a unit cross-sectional area MKS units J/m^2 .

ENRICHED MATERIAL: (1) Material in which the relative amount of one or more isotopes of a constituent has been increased. (2) Uranium in which the abundance of the isotope is increased above normal.

ENZYME: A biological catalyst of great specificity for a particular substance (substrate) or a particular group of closely related substances which generally activates or accelerates a chemical reaction.

EPIDERMIS: The outermost layer of cells of the skin.

EPILTION (DEPILATION): The temporary or permanent removal or loss of hair.

EPITHELIUM: A term applied to cells that line all canals and surfaces having communication with external air; also, cells specialized for secretion in certain glands as the liver, kidneys, etc.

EQUILIBRIUM, RADIOACTIVE: In a radioactive series, the state which prevails when the ratios between the amounts of successive members of the series remains constant.

SECULAR EQUILIBRIUM: If a parent element has a very much longer half-life than the daughters (so there is no appreciable change in its amount in the time interval required for later products to attain equilibrium) then, after equilibrium is reached, equal numbers of atoms of all members of the series disintegrate in unit time. This condition is never actually attained, but is essentially established in such a case as radium and its series to Radium D. The half-life of radium is about 1,600 years; of radon, approximately 3.82 days, and of each of the subsequent members, a few minutes. After about a month, essentially the equilibrium amount of radon is present; then (and for a long time) all members of the series disintegrate the same number of atoms per unit time.

TRANSIENT EQUILIBRIUM: If the half-life of the parent is short enough so the quantity present decreases appreciably during the period under consideration, but is still longer than that of successive members of the series, a stage of equilibrium will be reached after which all members of the series decrease in amount

GLOSSARY

exponentially with the period of the parent. An example of this is radon (half-life of approximately 3.82 days) and successive members of the series to Radium D.

ERG: Unit of work done by a force of one dyne through a distance of one cm; cgs units: dyne-cm or gm-cm²/sec².

ERROR, STATISTICAL: Errors in counting due to the random time-distributions of disintegrations.

ERYTHEMA: An abnormal redness of the skin due to distension of the capillaries with blood. It can be caused by many different agents heat, drugs, ultra-violet rays, ionizing radiation.

EXCITATION: The addition of energy to a system, thereby transferring it from its ground state to an excited state. Excitation of a nucleus, an atom, or a molecule can result from absorption of photons or from inelastic collisions with other particles.

EXCITED STATE: An energy level higher than the ground state energy level.

EXOERGIC: That which liberates energy.

EXPOSURE: A measure of the ionization produced in air by X or gamma radiation. It is the sum of the electrical charges on all ions of one sign produced in air when all electrons liberated by protons in a volume element of air are completely stopped in air, divided by the mass of the air in the volume element. The unit of exposure is the roentgen.

EXTERNAL EXPOSURE: Exposure to ionizing radiations coming from a source outside the body.

-F-

FALLOUT: Radioactive debris from a nuclear detonation, which is airborne or has been deposited on earth. Special forms of fallout are "Dry Fallout," "Rainout," and "Snowout."

FERTILE: Of a nuclide, capable of being transformed, directly or indirectly, into a fissile nuclide by neutron capture. Of a material, containing one or more fertile nuclides.

FILM BADGE: A pack of photographic film which measures radiation exposure for personnel monitoring. The badge may contain two or three films of differing sensitivity and filters to shield parts of the film from certain types of radiation.

FILM RING: A film badge in the form of a finger ring.

FILTER (RADIOLOGY): Primary - A sheet of material, usually metal, placed in a beam of radiation to absorb preferentially the less penetrating components. Secondary - A sheet of material of low atomic number (relative to the primary filter) placed in the filtered

GLOSSARY

beam of radiation to remove characteristic radiation produced by the primary filter.

FILTRATION, INHERENT (X-RAYS): The filter permanently in the useful beam; it includes the window of the X-ray tube and any permanent tube or source enclosure.

FISSILE: Of a nuclide, capable of undergoing fission by interaction with slow neutrons.

FISSION, NUCLEAR: A nuclear transformation characterized by the splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy.

FISSION PRODUCTS: Elements or compounds resulting from fission.

FISSION YIELD: The percentage of fissions leading to a particular nuclide.

FISSIONABLE: Of a nuclide, capable of undergoing fission by any process.

FLUENCE: The number of particles passing through a unit cross-sectional area.

FLUORESCENCE: The emission of radiation of particular wavelengths by a substance as a result of absorption of radiation of shorter wavelength. This emission occurs essentially only during the irradiation.

FLUOROGRAPHY (PHOTOFLUROGRAPHY): Photography of image produced on fluorescent screen by X or gamma radiation.

FLUOROSCOPE: A fluoroscent screen, suitably mounted with respect to an X-ray tube for ease of observation and protection, used for indirect visualization (by X rays) of internal organs in the body or internal structures in apparatus or in masses of material.

FLUX DENSITY (FLUENCE RATE): The number of particles passing through a unit cross-sectional area per unit of time. (Fluence per unit of time.)

FLUX, NEUTRON: A term used to express the intensity of neutron radiation. The number of neutrons passing through a unit area in unit time. For neutrons of a given energy, the product of neutron density with speed.

FOCAL SPOT (X RAYS): The part of the target of the X-ray tube struck by the main electron stream.

FUEL: Fissionable material of reasonably long life, used or usable in producing energy in a nuclear reactor. The term frequently is applied to a mixture, such as natural uranium, in which only part of the atoms are fissionable, if it can maintain a self-sustaining chain reaction under the proper conditions. **FUEL CYCLE:** The sequence of steps, such as utilization, reprocessing, and refabrication, through which nuclear fuel passes.

GLOSSARY

FUSION, NUCLEAR: Act of coalescing two or more atomic nuclei.

-G-

GAMMA, PROMPT: Gamma radiation emitted at the time of fission of a nucleus.

GAMMA RAY: Short wavelength electromagnetic radiation of nuclear origin (range of energy from 10 keV to 9 MeV) emitted from the nucleus.

GAS AMPLIFICATION: As applied to gas ionization radiation detecting instruments, the ratio of the charge collected to the charge produced by the initial ionizing event.

GEIGER COUNTER: A detector designed to detect ionizing particles. It usually consists of a conducting cylindrical tube with a wire running down its axis. A voltage is applied between the cylinder and the wire such that the ions produced by the charged particle in passing through the counter will start an electrical breakdown in the gas in the counter resulting in an electrical pulse which can be recorded.

GEIGER REGION: In an ionization radiation detector, the operating voltage interval in which the charge collected per ionizing event is essentially independent of the number of primary ions produced in the initial ionizing event.

GEIGER THRESHOLD: The lowest voltage applied to counter tube for which the number of pulses produced in the counter tube is essentially the same, regardless of a limited voltage increase.

GENERATOR ("COW"): A device in which a daughter radionuclide is eluted from an ion exchange column containing a parent radionuclide long-lived compared to the daughter.

GENETIC EFFECT OF RADIATION: Inheritable change, chiefly mutations, produced by the absorption of ionizing radiations. On the basis of present knowledge these effects are purely additive; there is no recovery.

GENETICS: The branch of biology dealing with the phenomena of heredity and variations.

GEOMETRY FACTOR: The fraction of the total solid angle about the source of radiation that is subtended by the face of the sensitive volume of a detector.

GEOMETRY, GOOD: In nuclear physics measurements, an arrangement of source and detecting equipment such that the use of finite source size and finite detector aperture introduces little error.

GEOMETRY, POOR: In a nuclear experiment, an arrangement in which the angular aperture between the source and detector is large,

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introducing into the measurement a comparative large uncertainty for which a correction may be necessary.

GRAM ATOMIC WEIGHT: A mass in grams numerically equal to the atomic weight of an element.

GRAM MOLECULAR WEIGHT (GRAM-MOLE): Mass in grams numerically equal to the molecular weight of a substance.

GRAM-RAD: Unit of integral dose equal to 100 ergs.

GRAPHITE: A form of carbon in which the atoms are hexagonally arranged in planes. Commonly used for moderators because it can be made in compact, fairly strong blocks, easily machined to close tolerances, and because the prolonged baking at high temperature used in its manufacture helps eliminate impurities that might absorb neutrons.

GRENZ RAYS: X rays produced at voltages of 5 to 20 kVp, intended primarily for surface therapy.

GRAY (Gy): A unit of absorbed dose. (One Gy = 1J/kg = 100 rads.)

GROUND STATE: The state of a nucleus, atom, or molecule at its lowest energy. All the other states are "excited."

-H-

HALF-LIFE, BIOLOGICAL: The time required for the body to eliminate one-half of an administered dosage of any substance by regular processes of elimination. Approximately the same for both stable and radioactive isotopes of a particular element.

HALF-LIFE, EFFECTIVE: Time required for a radioactive element in an animal body to be diminished 50 percent as a result of the combined action of radioactive decay and biological elimination. Effective half-life = {Biological half-life x Radioactive half-life}/{Biological half-life + Radioactive half-life}.

HALF-LIFE, RADIOACTIVE: Time required for a radioactive substance to lose 50 percent of its activity by decay. Each radionuclide has a unique half-life.

HALF VALUE LAYER (HALF THICKNESS) (HVL): The thickness of a given material which will absorb half the gamma radiation incident upon it. This thickness depends on the nature of the material - it is roughly inversely proportional to its density - and also on the energy of the gamma rays.

HARDNESS (X RAYS): A relative specification of the quality or penetrating power of x rays. In general, the shorter the wavelength the harder the radiation.

HEALTH PHYSICS: A term in common use for that branch of radiological physics dealing with the protection of personnel from the harmful

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effects of ionizing radiation. It includes the routine procedures of radiation protection surveys, area and personnel monitoring, the recommendation of appropriate protective equipment and procedures, the determination of acceptable standards of operation, and the solutions of problems incident to the effective and practical protection of all persons from harmful effects of radiation.

HEALTH, RADIOLOGICAL: The art and science of protecting human beings from injury by radiation, and promoting better health through beneficial applications of radiation.

HEREDITY: Transmissions of characters and traits from parent to offspring.

HERTZ: Unit of frequency equal to one cycle per second.

HOT CELL: A heavily shielded enclosure for handling and processing (by remote means or automatically) or storing highly radioactive materials.

HOT SPOT: Region in a contaminated area in which the level of radioactive contamination is somewhat greater than in neighboring regions in the area.

-I-

IMMUNITY: The power which a living organism possesses to resist and overcome infection.

IMPLANT (RADIOLOGY): Encapsulated radioactive material embedded in a tissue for therapy. It may be permanent (seed) or temporary (needle).

INITIAL NUCLEAR RADIATION: Nuclear radiation (essentially neutrons and gamma rays) emitted from the ball of fire and the cloud column during the first minute after a nuclear (or atomic) explosion. The time limit of one minute is set, somewhat arbitrarily, as that required for the source of the radiations (fission products in the atomic cloud) to attain such a height that only insignificant amounts reach the earth's surface.

INSTRUMENT RANGE: The region between the limits within which a quantity is measured, received, or transmitted, expressed by stating the lower and upper range values. In a nuclear reactor, the range of operating power levels in which the power is too low to manifest itself in terms of measurable temperature changes and must be measured by nuclear instruments, such as ionization chambers; usually considered to involve power levels representing a multiplication of the startup source by a factor of 100 or more thereby distinguished from the lower "source range."

INTENSIFYING SCREEN: Sheet of cardboard or other substance coated with fluorescent material, placed in contact with the film in radiography. The X or gamma rays excite the fluorescent substance. The light thus emitted adds to the radiation effect on the film and

GLOSSARY

produces an image of greater density for a given exposure. Sheets of thin lead may be used in industrial radiography with very high energy radiation. In this case, the increased effect is due largely to secondary electrons and x rays emitted by the lead.

INTENSITY: Amount of energy per unit time passing through a unit area perpendicular to the line of propagation at the point in question.

INTERLOCK: A device, usually electrical and (or) mechanical, to prevent activation of a control until a preliminary condition has been met, or prevent hazardous operations. Its purpose usually is safety.

INTERNAL CONTAMINATION EXPOSURE: The radiation exposure resulting from the deposition of radioactive materials within body tissues. A continuous, and often long-term, hazard results from the deposition of alpha, beta, and (or) gamma emitters.

INTERNAL CONVERSION: One of the possible mechanisms of decay from the metastable state (isomeric transition) in which the transition energy is transferred to an orbital electron, causing its ejection from the atom. The ratio of the number of internal conversion electrons to the number of gamma quanta emitted in the de-excitation of the nucleus is called the "conversion ratio."

ION: Atomic particle, atom, or chemical radical bearing an electrical charge, either negative or positive.

ION EXCHANGE: A chemical process involving the reversible interchange of ions between a solution and a particular solid material such as an ion exchange resin consisting of a matrix of insoluble material interspersed with fixed ions of opposite charge.

IONIZATION: The process by which a neutral atom or molecule acquires a positive or negative charge.

PRIMARY IONIZATION: (1) In collision theory: the ionization produced by the primary particles as contrasted to the "total ionization" which includes the "secondary ionization" produced by delta rays. (2) In counter tubes: the total ionization produced by incident radiation without gas amplification.

SECONDARY IONIZATION: Ionization produced by delta rays.

SPECIFIC IONIZATION: Number of ion pairs created per unit length of path ionizing radiation in a medium; e.g., per cm of air or per micron of tissue.

TOTAL IONIZATION: The total electric charge of one sign on the ions produced by radiation in the process of losing its kinetic energy. For a given gas, the total ionization is closely proportional to the initial ionization radiation. It is frequently used as a measure of radiation energy.

IONIZATION CHAMBER: An enclosure connecting two or more electrodes

GLOSSARY

between which an electric field is maintained to collect the charge when the enclosed gas is ionized. The ions formed and registered are a measure of the incident radiation which has caused the ionization.

IONIZATION DENSITY: Number of ion pairs per unit volume.

IONIZATION PATH (TRACK): The trail of ion pair produced by an ionizing radiation in its passage through matter.

IONIZING EVENT: Any occurrence of a process in which an ion or group of ions is produced.

ION PAIR: Two particles of opposite charge, usually referring to the electron and positive atomic or molecular residue resulting after the interaction of ionizing radiation with the orbital electrons of atoms.

IRRADIATION: Exposure to radiation.

ISOBARS: Nuclides having the same mass number but different atomic numbers.

ISOCRONOUS CYCLOTRON (AZIMUTHALLY VARYING FIELD {AVF} OR SECTOR FOCUSED CYCLOTRON): A cyclotron which uses a constant accelerating frequency and focuses the particles by means of wedge-shaped sectors on the magnet poles.

ISODOSE CHART: Chart showing the distribution of radiation in a medium by means of lines or surfaces drawn through points receiving equal doses. Isodose charts have been determined for beams of X-rays traversing the body, for radium applicators used for intra cavitary or interstitial therapy, and for working areas where X-rays or radioactive nuclides are employed.

ISODOSE CURVE: A curve depicting loci of identical radiation doses in a structure.

ISOMETRIC TRANSITION: The process by which a nuclide decays to an isomeric nuclide (i.e., one of the same mass number and atomic number) and lower quantum energy. Isomeric transitions (often abbreviated I.T.) proceed by gamma ray and (or) internal conversion electron emission.

ISOMERS: Nuclides having the same number of neutrons and protons but capable of existing, for a measureable time, in different quantum states with different energies and radioactive properties. Commonly, the isomer of higher energy decays to one with lower energy by the process of isomeric transition.

ISOTONES: Nuclides having the same number of neutrons in their nuclei.

ISOTOPES: Nuclides having the same number of protons in their nuclei, and hence the same atomic number, but differing in the number of neutrons, and therefore in the mass number. Almost identical chemical properties exist between isotopes of a particular element. The term should not be used as a synonym for nuclide.

GLOSSARY

STABLE ISOTOPE: A non-radioactive isotope of an element.

ISOTOPE EFFECT (CHEMISTRY): The effect in the difference in the mass between isotopes of an element on the rate and (or) equilibria of chemical transformation.

ISOTOPE SEPARATION: Process in which a mixture of isotopes of an element is separated into its component isotopes, or in which the abundance of isotopes in such a mixture is changed.

-J-

JOULE: The unit for work and energy, equal to one newton expended along the distance of one meter ($1 \text{ J} = 1 \text{ N} \times 1 \text{ m}$).

-K-

K-CAPTURE (ELECTRON CAPTURE): Electron capture from the K-shell by the nucleus of the atom.

KERMA: The sum of the initial kinetic energies of all charged particles liberated by indirectly ionizing particles in a volume, divided by the mass of number in that volume.

KILO ELECTRON VOLT (keV): One thousand electron volts, 10^3 eV.

KILOVOLT (kV): A unit of electrical potential difference equal to 1,000 volts.

KILOVOLT PEAK (kVp): The peak value in kilovolts of the potential difference of a pulsating potential generator. When only half the wave is used, the value refers to the useful half of the cycle.

KLEIN-NISHINA FORMULA: A formula that expresses the cross section of an unbound electron for scattering of a photon in the Compton effect, as a function of the energy of the photon. The term usually refers to the integral Klein-Nishina formula which gives the total cross section for the process. The differential Klein-Nishina formula gives the differential cross section for scattering at a given angle. Because of the confidence with which photo-electron interactions can be interpreted (by using the Klein-Nishina formula), the Compton effect is important in the analysis of energy and polarization of gamma rays from many sources.

-L-

LABELED COMPOUND: A compound consisting, in part, of labeled molecules. By observations of radioactivity or isotopic composition, this compound or its fragments may be followed through physical, chemical, or biological processes.

GLOSSARY

LABELED MOLECULE: A molecule containing one or more atoms distinguished by non-natural isotopic composition (with radioactive or stable isotopes).

LAG TIME: The time between the occurrence of the primary ionizing event and the occurrence of the count.

LATENT PERIOD: The period or state of seeming inactivity between the time of exposure of tissue to an injurious agent and response.

LD (RADIATION DOSE): (see DOSE, MEDIAN LETHAL).

LEAD EQUIVALENT: The thickness of lead affording the same attenuation, under specific conditions, as the material in question.

LEPTON: One of a class of light elementary particles (having small mass). Specifically, an electron, a positron, a neutrino, an antineutrino, a muon, or an antimuon.

LEUKEMIA: A disease in which there is a great overproduction of white blood cells, or a relative overproduction of immature white cells, and great enlargement of the spleen. The disease is variable, at times running a more chronic course in adults than in children. It is almost always fatal. It can be produced in some animals by long-continued exposure to low doses of ionizing radiation.

LINEAR ACCELERATOR: A device for accelerating charge particles. It employs alternate electrodes and gaps arranged in a straight line, so proportioned that when potentials are varied in the proper amplitude and frequency, particles passing through the wavelength receive successive increments of energy.

LINEAR ENERGY TRANSFER (LET): The average energy locally imparted to a medium by a charged particle of specified energy per unit distance traversed.

LOCALIZATION, SELECTIVE (BIOLOGY): Accumulation of particular nuclide to a significantly greater degree in certain cells or tissue. (see ABSORPTION RATIO, DIFFERENTIAL).

-M-

MASS (CRITICAL): The minimum mass of fissile material which can be made critical with a specified geometrical arrangement and material composition.

MASS (RELATIVISTIC): The increased mass associated with a particle when its velocity is increased. The increase in mass becomes appreciable only at velocities approaching the velocity of light, 3×10^{10} cm/sec.

MASS DEFECT: Difference between the mass of the nucleus as a whole and the sum of the component nucleon masses.

GLOSSARY

MASS-ENERGY RELATION: The name sometimes given to the equation $E = mc^2$.

MASS NUMBERS: The number of nucleons (protons and neutrons) in the nucleus of an atom. (Symbol: A.)

MAXIMUM CREDIBLE ACCIDENT: The worst accident in a reactor or nuclear energy installation that, by agreement, need be taken into account in devising protective measures.

MAXIMUM PERMISSIBLE CONCENTRATION (MPC): The highest currently acceptable concentration of radioactive substances (usually expressed as microcuries per cubic centimeter) in air, water, or food to which an individual may be exposed throughout a stated period of time, without expectation of injury.

MAXIMUM PERMISSIBLE EXPOSURE (MPE): The total amount of radiation exposure to which a normal person may be subjected day-by-day without any harmful effects becoming evident during his lifetime.

MAXIMUM PERMISSIBLE LIMIT (MPL): The total amount of internal contamination and external radiation combined that will not give a dose that will produce appreciable bodily injury. The term MPL refers to the MPC and MPE collectively and is the sum of the dose from external and internal sources. ($MPL = MPE + MPC$) The limits on the dose rates and surface contamination levels necessary to assure compliance with the established total permissible dose are also called MPL's.

MEAN FREE PATH: The average distance that particles of a specified type travel before a specified type (or types) of interaction in a given medium. The mean free path may thus be specified for all interactions (i.e., total mean free path) or for particular types of interaction such as scattering, capture, or ionization.

MEAN LIFE: The average lifetime for an atomic or nuclear system in a specified state. For an exponentially decaying system, the average time for the number of atoms or nuclei in a specified state to decrease by a factor of e (2.718...).

MEGA ELECTRON VOLT (MeV): One million electron volts, 10^6 eV.

MESON: One of a class of medium-mass, short-lived elementary particles with a mass between that of the electron and that of the proton. Examples: Pimesons (pions) and K-mesons (kaons).

METABOLISM: The sum of all physical and chemical processes by which living organized substance is produced and maintained and by which energy is made available for the uses of the organism.

METASTABLE STATE: An excited nuclear state having a half-life long enough to be observed.

METASTASIS: The transfer in the body of malignant neoplastic cells from the original or parent site to one more distant.

MeV: The symbol for one million electron volts, or 10^6 eV (written

GLOSSARY

MeV).

MICRON: Unit of length equal to 10^{-6} meters.

MICROWAVE: An electromagnetic wave having a wavelength of approximately 1 meter to 1 millimeter corresponding to frequencies of about 300 to 300,000 megacycles per second.

MIL: Unit of length equal to one-thousandth of an inch.

MILLIROENTGEN (mR): A submultiple of the roentgen, equal to one one-thousandth of a roentgen. (see ROENTGEN.)

MODERATOR: Material used to moderate or slow down neutrons from the high energies at which they are released.

MOLECULAR WEIGHT: The sum of the atomic weights of all the atoms in a molecule.

MOLECULE: Smallest quantity of a compound which can exist by itself and retain all properties of the original substance.

MOMENTUM: The product of the mass of a body and its velocity; MKSA units, kg-m/sec.

MONTE-CARLO METHOD: A method permitting the solution by means of a computer of problems of physics, such as those of neutron transport, by determining the history of a large number of elementary events by the application of the mathematical theory of random variables.

MONITORING: Periodic or continuous determination of the amount of ionizing radiation or radioactive contamination present in an occupied region.

AREA MONITORING: Routine monitoring of the radiation level or contamination of a particular area, building, room, or equipment. Some laboratories or operations distinguish between routine monitoring and survey activities.

PERSONNEL MONITORING: Monitoring any part of the individual, his breath, or excretions, or any part of his clothing.

MULTIPLICATION FACTOR (k): The ratio of the total number of neutrons produced during a time interval (excluding neutrons produced by sources whose strengths are not a function of fission rate) to the total number of neutrons lost by absorption and leakage during the same interval. When the quantity is evaluated for an infinite medium or for an infinite repeating lattice, it is referred to as the infinite multiplication factor (k_k) and when the quantity is evaluated for a finite medium, it is referred to as the effective multiplication factor (k_{eff}).

MUTATION: Alteration of the usual hereditary pattern, usually sudden.

-N-

GLOSSARY

N-UNIT: That quality of neutron radiation measured in a condensor R-meter that will produce the same amount of ionization as one roentgen of X radiation.

NEOPLASM: A new growth of cells which is more or less unrestrained and not governed by the usual limitations of normal reproduction. Benign: some degree of growth restraint and no spread to distant parts. Malignant: growth invades tissues or spreads to distant parts, or both.

NEUTRINO: Elementary nuclear particle with a mass approximately the same as that of a hydrogen atom and electrically neutral; its mass is 1.008982 mass units. Neutrons are commonly divided into sub-classifications according to their energies as follows: thermal, around 0.025 eV; epithermal, 0.1 eV to 100 eV; slow, less than 100 eV; intermediate, 10^2 to 10^5 eV; fast, greater than 0.1 MeV.

NEUTRON-INDUCED ACTIVITY: Radioactivity produced in a nucleus from neutron capture.

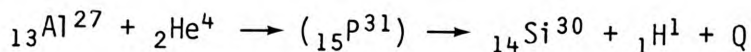
NEUTRON CYCLE: The average energy, interaction and migration history of neutrons in a reactor, beginning with fission and continuing until they have leaked out or have been absorbed.

NEUTRONS, PROMT: Neutrons accompanying the fission process without measurable delay.

NEWTON: The unit of force, which when applied to a one kilogram mass will give it an acceleration of one meter per second.

NUCLEAR FUSION: (See REACTION, THERMONUCLEAR)

NUCLEAR REACTION: A process in which a particle hits a nucleus, often forming a compound nucleus which can decay by giving off the same particle with the same or less energy or by giving off another particle. The resultant or product nucleus is sometimes radioactive. The notation sometimes used to describe nuclear reactions is illustrated below for the case of an alpha particle interacting with an aluminum nucleus resulting in a proton being given off.



This is often abbreviated to ${}_{13}\text{Al}^{27} (\alpha, p) {}_{14}\text{Si}^{30}$. The number to the lower left of the chemical symbol is the atomic number and the number to the upper right of the symbol is the mass number. Q represents the Q value or energy balance. The parentheses enclose the compound nucleus.

NUCLEON: Common name for a constituent particle of the nucleus. Applied to the proton or a neutron.

NUCLEUS: (Biological) A definitely delineated body within the cell, containing the chromosomes. (Nuclear) That part of an atom in which the total positive electric charge and most of the mass is

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concentrated.

NUCLIDE: A species of atom characterised by the constitution of its nucleus. The nuclear constitution is specified by the number of protons (Z), number of neutrons (N), and energy content; or, alternatively, by the atom number (Z), mass number $A = (N + Z)$, and atomic mass. To be regarded as a distinct nuclide, the atom must be capable of existing for a measurable time. Thus, nuclear isomers are separate nuclides, whereas promptly decaying excited nuclear reactions are not so considered.

-O-

ORGAN: Group of tissues which together perform one or more definite functions in a living body.

OVERMODERATED: Of a multiplying system, having a moderator-to-fuel volume ratio greater than that which makes for some specified reactor parameter an optimum value.

-P-

PACKING FRACTION: The ratio ($\Delta M/A$) of the mass defect (ΔM), and mass number (A), of a nuclide.

PAIR PRODUCTION: An absorption process for X and gamma radiation in which the incident photon is annihilated in the vicinity of the nucleus of the absorbing atom, with subsequent production of an electron and positron pair. This reaction only occurs for incident photon energies exceeding 1.02 MeV.

PARENT: A radionuclide which, upon disintegration, yields a specified nuclide—either directly or as a later member of a radioactive series.

PATH, MEAN FREE: Average distance a particle travels between collisions.

PERIODIC TABLE: An arrangement of chemical elements in order of increasing atomic number. Elements of similar properties are placed one under the other, yielding groups and families of elements. Within each group there is a gradation of chemical and physical properties but, in general, a similarity of chemical behavior. From group to group, however, there is a progressive shift of chemical behavior from one end of the table to the other.

PHANTOM: A volume of material approximately as closely as possible the density and effective atomic number of tissue. Ideally a phantom should behave in respect to absorption of radiation in the same manner as tissue. Radiation dose measurements made within or on a phantom provide a means of determining the radiation dose within or on a body under similar exposure conditions. Some materials commonly used in phantoms are water, Masonite, pressed wood, and beeswax.

GLOSSARY

PHOSPHORESCENCE: Emission of radiation by a substance as a result of previous absorption of radiation of shorter wavelength. In contrast to fluorescence, the emission may continue for a considerable time after cessation of the exciting irradiation.

PHOTOELECTRIC EFFECT: Process by which a photon ejects an electron from an atom. All the energy of the photon is absorbed in ejecting the electron and in imparting kinetic energy to it.

PHOTOFLUOROGRAPHY: (See FLUOROGRAPHY)

PHOTOMULTIPLIER TUBE: A vacuum tube which converts light signals into electrical pulses. The tube consists of a light sensitive surface or photocathode which give off photoelectrons when light is incident on it. These electrons are directed and focused using electric fields so that they strike perhaps 10 or 14 successive stages or dynodes at each of which the number of electrons are multiplied by a process called secondary emission. The result is that a single photon incident on the photocathode can result in several million electrons at the other end of the tube which can produce electrical pulses big enough to be amplified and recorded. Such tubes are very useful in recording the scintillations produced by charged particles in passing through scintillating materials such as sodium iodide crystals. They are thus usually an essential part of a modern scintillation counter.

PHOTON: A quantity of electromagnetic energy (E) whose value in joules is the product of its frequency (ν) in hertz and Planck constant (h). The equation is: $E = h\nu$.

PILE: (See REACTOR, NUCLEAR)

PLANCK CONSTANT: A natural constant of proportionality (h) relating the frequency of a quantum of energy to the total energy of the quantum:

$$h = E/\nu = 6.6256 \times 10^{-34} \text{ joule-seconds}$$

PLATEAU: As applied to radiation detection chambers, the level portion of the counting rate-voltage curve where changes in operating voltage introduce minimum changes in the counting rate.

PLATEAU SLOPE, RELATIVE: The relative increase in the number of counts as a function of voltage expressed in percentage per 100 volts increase above the Geiger threshold.

POISON: Material of high absorption cross section which absorbs neutrons unproductively and reduces the reactivity of a reactor.

POLYMERIZATION: Union of two or more molecules of a compound to form a more complex molecule.

POSITRON: Particle equal in mass to the electron and having an equal but positive charge.

POTENTIAL, IONIZATION: The potential necessary to separate one electron from an atom, resulting in the formation of an ion pair.

GLOSSARY

POTENTIAL DIFFERENCE: Work required to carry a unit positive charge from one point to another.

POWER, NUCLEAR: Useful power released in exothermic nuclear reactions.

POWER, STOPPING: A measure of the effect of a substance upon the kinetic energy of a charged particle passing through it.

PRESSURE VESSEL, REACTOR: A reactor vessel designed to withstand a substantial operating pressure.

PROMPT GAMMA RADIATION: Gamma radiation accompanying the fission process without measurable delay.

PROPORTIONAL COUNTER: A gas filled radiation counter designed to count charged particles. Often built much like a Geiger counter in the form of a cylinder with a wire running along its axis. A voltage, less than in the Geiger counter is applied between the cylinder and the wire such that only a few ions are produced resulting in a much smaller electrical pulse from the counter but one which is proportional to the energy of the particle producing it.

PROPORTIONAL REGION: Voltage range in which the gas amplification is greater than one, and in which the charge collected is proportional to the charge produced by the initial ionizing event.

PROTIUM: A name sometimes applied to the hydrogen isotope of mass 1 to distinguish it from deuterium and tritium.

PROTON: Elementary nuclear particle with a positive electric charge equal numerically to the charge of the electron and a mass of 1.007277 mass units.

PULSE HEIGHT SELECTOR: A circuit designed to select and pass voltage pulses of a certain minimum amplitude.

-Q-

QUALITY (RADIOLOGY): The characteristic spectral-energy distribution of X radiation. It is usually expressed in terms of effective wavelengths or half-value layers of a suitable material; e.g., up to 20 kV, cellophane; 20 to 120 kVp, aluminum; 120 to 400 kVp, copper; over 400 kVp, tin.

QUALITY FACTOR (QF): The linear-energy-transfer-dependent factor by which absorbed doses are multiplied to obtain (for radiation protection purposes) a quantity that expresses - on a common scale for all ionizing radiations - the effectiveness of the absorbed dose.

QUANTUM: An observable quantity is said to be "quantized" when its magnitude is, in some or all of its range, restricted to a discrete set of values. If the magnitude of the quantity is always a multiple of a definite unit, then that unit is called the quantum (of the

GLOSSARY

quantity). For example, the quantum or unit of orbital angular momentum is h , and the quantum of energy of electromagnetic radiation of frequency ν is $h\nu$. In field theories, a field (or the field equations) is quantized by application of a proper quantum-mechanical procedure. This results in the existence of a fundamental field particle, which may be called the field quantum. Thus, the photon is a quantum of the electromagnetic field and in nuclear field theories the meson is considered the quantum of the nuclear field.

QUANTUM THEORY: The concept that energy is radiated intermittently in units of definite magnitude called quanta, and absorbed in a like manner.

QUENCHING: The process of inhibiting continuous or multiple discharge in a counter tube which uses gas amplification.

QUENCHING VAPOR: Polyatomic gas used in Geiger-Mueller counters to quench or extinguish ionization.

-R-

RABBIT: A small container propelled, usually pneumatically or hydraulically, through a tube in a nuclear reactor to expose substances experimentally to the radiation and neutron flux of the active section. Used for rapid removal of samples with very short half-lives.

RAD: The unit of absorbed dose equal to 0.01 J/kg in any medium. (See ABSORBED DOSE.) (Written: rad.)

RADIAC: A term referring to radiation-measuring instruments (dosimeters, dose rate meters, laboratory counters and scalers, alpha survey meters, etc.). The term comes from Radioactivity Detection, Indication, and Computation. Radiac may be used as a noun (e.g., beta-gamma radiacs) or as an adjective (e.g., beta-gamma radiac equipment).

RADIATION: (1) The emission and propagation of energy through space or through a material medium in the form of waves; for instance, the emission and propagation of electromagnetic waves, or of sound and elastic waves. (2) The energy propagated through space or through a material medium as waves; for example, energy in the form of electromagnetic waves or of elastic waves. The term radiation or radiant energy, when unqualified usually refers to electromagnetic radiation. Such radiation commonly is classified, according to frequency, as Hertzian, infrared, visible (light), ultra-violet, X-ray, and gamma ray. (See PHOTON.) (3) By extension, corpuscular emissions, such as alpha and beta radiation, or rays of mixed or unknown type, as cosmic radiation.

ANNIHILATION RADIATION: Photons produced when an electron and a positron unite and cease to exist. The annihilation of a positron-electron pair results in the production of two photons, each of 0.51 MeV energy.

GLOSSARY

BACKGROUND RADIATION: Radiation arising from radioactive material other than the one directly under consideration. Background radiation due to cosmic rays and natural radioactivity is always present. There may also be background radiation due to the presence of radioactive substances in other parts of the building, in the building material itself, etc.

CHARACTERISTIC (DISCRETE) RADIATION: Radiation originating from an atom after removal of an electron or excitation to the nucleus. The wavelength of the emitted radiation is specific, depending only on the nuclide and particular energy levels involved.

DIRECT RADIATION: Obsolete term for "leakage radiation."

EXTERNAL RADIATION: Radiation from a source outside the body - the radiation must penetrate the skin.

INFRARED RADIATION: Invisible thermal radiation whose wavelength is longer than the red segment of the visible spectrum.

INTERNAL RADIATION: Radiation from a source within the body (as a result of deposition of radionuclides in the body tissues).

IONIZING RADIATION: Any electromagnetic or particulate radiation capable of producing ions, directly or indirectly, in its passage through matter.

LEAKAGE (DIRECT) RADIATION: All radiation coming from the source housing except the useful beam.

MONOCHROMATIC RADIATION: Electromagnetic radiation of a single wavelength, or radiation in which all the photons have the same energy.

MONOENERGETIC RADIATION: Radiation of a given type (alpha, beta, neutron, gamma, etc.) in which all particles or photons originate with and have the same energy.

NUCLEAR RADIATION: Particulate and electromagnetic radiation emitted from atomic nuclei in various nuclear processes. The important nuclear radiations are alpha and beta particles, gamma rays, and neutrons. All nuclear radiations are ionizing radiations, but the reverse is not true; X rays, for example, are included among ionizing radiations since they do not originate from atomic nuclei.

PRIMARY RADIATION: The useful beam of an X-ray tube.

PROMPT RADIATION: All ionizing radiation emitted during the fission and fusion processes of a nuclear weapon. (See INITIAL NUCLEAR RADIATION.)

RESIDUAL NUCLEAR RADIATION: Nuclear radiation, chiefly beta particles and gamma rays, which persists for some time following a nuclear (or atomic) explosion. The radiation is emitted

GLOSSARY

mainly by the fission products and other bomb residues in the fallout, and to some extent by earth and water constituents, and other materials in which radioactivity has been induced by the capture of neutrons.

SCATTERED RADIATION: Radiation which during its passage through a substance, has been deviated in direction. It may also have been modified by a decrease in energy.

SECONDARY RADIATION: Radiation resulting from absorption of other radiation in matter. It may be either electromagnetic or particulate.

STEM RADIATION: X rays given off from parts of the anode other than the target, particularly from the target support.

STRAY RADIATION: The sum of leakage and scattered radiation.

RADIOACTIVE EQUILIBRIUM: The state which prevails when the ratios between the amounts of successive members of a radioactive decay series remain constant.

RADIOACTIVITY: The property of certain nuclides of spontaneously emitting particles or gamma radiation or of emitting X radiation following orbital electron capture or of undergoing spontaneous fission.

ARTIFICIAL RADIOACTIVITY: Man-made radioactivity produced by particle bombardment or electromagnetic irradiation, as opposed to natural radioactivity.

INDUCED RADIOACTIVITY: Radioactivity produced in a substance after bombardment with neutrons or other particles. The resulting activity is "natural radioactivity" if formed by nuclear reactions occurring in nature, and "artificial radioactivity" if the reactions are caused by man.

NATURAL RADIOACTIVITY: The property of radioactivity exhibited by more than fifty naturally occurring radionuclides.

RADIOBIOLOGY: That branch of biology which deals with the effects of radiation on biological systems.

RADIOCHEMISTRY: The aspects of chemistry with radionuclides and their properties, with the behavior of minute quantities of radioactive materials by means of their radioactivity, and the use of radionuclides in the study of chemical problems.

RADIOGRAPHY: The making of shadow images on photographic emulsion by the action of ionizing radiation. The image is the result of the differential attenuation of the radiation in its passage through the object being radiographed.

RADIOISOTOPE: An unstable nuclide which emits particulate and(or) electromagnetic radiation in the process of transforming to a stable state.

GLOSSARY

RADIOLOGY: That branch of medicine which deals with the diagnostic and therapeutic applications of radiant energy including X rays and radionuclides.

RADIOPHARMACEUTICAL: A pharmaceutical compound which has been tagged with a radionuclide.

RADIORESISTANCE: Relative resistance of cells, tissues, organs, or organisms to the injurious action of radiation. The term may also be applied to chemical compounds or to any substances. (See **RADIOSENSITIVITY**.)

RADIOSENSITIVITY: Relative susceptibility of cells, tissues, organs, organisms, or any living substance to the injurious action of radiation. Radioresistance and radiosensitivity are currently used in a comparative sense, rather than in an absolute one.

RANGE: The range of a charged particle in a given substance is the distance that the particle will travel before being brought to rest as the result of ionization losses. The range depends upon the energy of the particle.

RATE, RECOVERY: The rate at which recovery takes place after radiation injury. It may proceed at different rates for different tissues. "Differential recovery rate." Among tissues recovering at different rates, those having slower rates will ultimately suffer greater damage from a series of successive irradiations. This differential effect is considered in fractionated radiation therapy if the neoplastic tissues have a slower recovery rate than surrounding normal structures.

RBE (RELATIVE BIOLOGICAL EFFECTIVENESS): The ratio of the number of rads of gamma (or X) radiation of a certain energy which will produce a specified biological effect to the number of rads of another radiation required to produce the same effect is the RBE of this latter radiation.

REACTION (NUCLEAR): An induced nuclear disintegration; i.e., a process occurring when a nucleus comes in contact with a photon, an elementary particle, or another nucleus. In many cases the reaction can be represented by the symbolic equation: $X + a \rightarrow Y + b$ or, in abbreviated form, $X(a,b)Y$. X is the target nucleus, a is the incident particle or photon, b is an emitted particle or photon, and Y is the product nucleus.

CHAIN REACTION: Any chemical or nuclear process in which some products or energy released by the process are instrumental in the continuation or magnification of the process.

ENDOERGIC REACTION: Reaction which absorbs energy.

ENDOTHERMIC REACTION: Reaction which absorbs energy, specifically in the form of heat.

EXOTHERMIC REACTION: Reaction which liberates energy, specifically in the form of heat.

GLOSSARY

THERMONUCLEAR REACTION: A nuclear reaction in which the energy necessary for the reaction is provided by colliding particles possessing kinetic energy by virtue of their thermal agitation. Such reactions occur at appreciable rates only for temperatures of millions of degrees and higher. Their rate increases with temperature. The energy of most stars is believed to be derived from exothermic thermonuclear reactions.

REACTIVITY: A parameter, ρ , giving the deviation from criticality of a nuclear chain-reacting medium such that positive values correspond to a supercritical state and negative values to a subcritical state.

REACTOR, BREEDER: A reactor which produces more fissile material than it consumes, i.e., has a conversion ratio greater than unity.

REACTOR, CONVERTER: A reactor which produces fissile atoms from fertile atoms, but has a conversion ratio less than one.

REACTOR, NUCLEAR: An apparatus in which nuclear fission may be sustained in a self-supporting chain reaction. A reactor includes fissionable material (fuel) such as uranium or plutonium, and moderating material (except fast reactors), and usually includes a reflector to conserve escaping neutrons, provision for heat removal, and measuring and control elements. The term "pile" and "reactor" have been used interchangeably, with reactor now becoming more common. These terms usually are applied only to systems in which the reaction proceeds as a controlled rate, but they also have been applied to bombs. Reactors may be classified on various bases:

1. By Fuel Arrangement

HETEROGENEOUS: Fissionable material, (fuel) and moderator are arranged as discrete bodies (usually in a regular pattern) of such dimensions that a non-homogeneous medium is presented to the neutrons.

HOMOGENEOUS: Fissionable material and moderator (if used) are so combined that an effectively homogeneous medium is presented to the neutrons. Such a mixture is represented either by a solution of fuel in moderator or by discrete particles whose dimensions are small in comparison with the neutron mean free path.

2. By Neutron Energy

EPITHERMAL: A substantial fraction of fissions (e.g., 30 or 40 percent) are induced by neutrons of more than thermal energy.

FAST: A nuclear reactor in which there is little moderation of neutrons. Thus, fission is induced primarily by fast neutrons that have lost relatively little of the energy with which they were released. The slowing down of neutrons that does occur is due largely to inelastic scattering instead of elastic scattering. About 100,000 electron volts is regarded as the minimum value of mean energy of neutrons inducing fission for a reactor to be considered fast, with one-half to one-third MeV

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more common. Sometimes the fission threshold of ^{238}U is taken as the lower limit of the fast range. Reactors of this type have potentially high neutron economy.

INTERMEDIATE: Fission is induced predominantly by neutrons whose energies are greater than thermal, but much less than the energy with which neutrons are released in fission. From 0.5 to 100,000 electron volts may be taken roughly as the energy range of neutrons inducing fission in intermediate reactors. The neutron absorption resonances of the fuel may be important in this range.

THERMAL: A nuclear reactor in which fission is induced primarily by neutrons of such energy that they are in substantial thermal equilibrium with the core material. A representative energy for thermal neutrons often is taken as 0.025 eV (2200 meters per second) which corresponds to the mean energy of neutrons in a Maxwellian distribution at 239 K, although most thermal reactors actually operate at a higher temperature. A moderator is an essential element of a thermal reactor.

3. By Use

POWER: A reactor capable of providing useful mechanical power. In reactors now planned, this is done by generating energy (in the form of heat) conveyed at a temperature high enough for efficient conversion to mechanical work.

POWER BREEDER: A nuclear reactor designed to produce both useful power and fuel.

PRODUCTION: A nuclear reactor designed primarily for large-scale production of transmutation products (e.g., plutonium).

RESEARCH: A reactor whose primary purpose is as a research tool. It may supply neutrons, other particles, and gamma radiation, and will include special provision for exposing samples (which may include living organisms) to these fluxes. It may provide transmutation products as well as have special experimental facilities.

HIGH FLUX: Since a high flux results from a high rate of fission per unit volume, a high-flux reactor operates at high power density.

HIGH TEMPERATURE: Roughly, the temperature may be considered high in this connection if it is great enough to permit generation of mechanical power at good efficiency.

RECOIL, AGGREGATE: The ejection, from the surface of a sample, of a cluster of atoms attached to one atom that is recoiled as the result of alpha particle emission. Although the phenomena may be quite common, the amount of matter thus carried away is so small as to be undetectable unless it is strongly radioactive. It is observed with strong preparations of alpha-active materials of high specific activity - such as nearly pure polonium compounds - as a migration of

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a small fraction of the radioactivity onto clean surfaces in the vicinity.

RECOMBINATION: The return of an ionized atom or molecule to the neutral state.

RECOVERY (RADIOBIOLOGY): The return toward normal of a particular cell, tissue, or organism after radiation injury.

REFLECTOR: A layer or structure of material between the shield and core of a reactor, designed to reduce the escape of neutrons and return them to the core. Neutrons entering the reflector are scattered randomly, some of them many times. A large fraction may ultimately return to the core. It is possible to design a reflector which will return more than 90 percent of the neutrons that would otherwise be lost. Requirements for a good reflector are similar to those for a good moderator: Its atoms should have low neutron-absorption cross section and high scattering cross section. Low atomic mass is not important. A reflector's effectiveness increases with its thickness, approaching a limiting factor when the thickness is several times the transport mean free path. Reflector savings is a measure of the decrease in critical core size obtained by the use of the reflector.

REM: A special unit of dose equivalent. The dose equivalent in rems is numerically equal to the absorbed dose in rads multiplied by the quality factor, the distribution factor, and any other necessary modifying factors.

REMOVABLE CONTAMINATION: Radioactive contamination which is attached to a surface and which may be removed by contact with skin or clothing, or that which may be removed by physical and (or) chemical means.

REP: An obsolete special unit of absorbed dose.

RESOLVING TIMES COINCIDENCE: The resolving time of a coincidence circuit is the time interval within which two or more pulses can arrive at the circuit and still be recorded as occurring at the same time. The term is also used with counting systems to specify the shortest time that can elapse between successive events and still have them counted. This can result from the fact that some counters are insensitive for a short time after registering a particle and from the fact that some recording equipment can only handle successive pulses separated by more than a certain minimum time interval.

RESOLVING TIME, COUNTER: The minimum time interval between two distinct events which will permit both to be counted. It may refer to an electronic circuit, to a mechanical indicating device, or to a counter tube.

RESONANCE ENERGY: The kinetic energy of an incident particle (expressed in the laboratory system) that makes the total energy of the system composed of the incident particle and the target nucleus close to the energy of a nuclear level of the compound nucleus.

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RESONANT TRANSFORMER: A transformer so designed that the inductance and distribution capacitance of its windings comprise a circuit which is in resonance at the frequency of the supplied power. As it does not require an iron core, insulation problems and weight are minimized. This principle is the basis of certain 1 and 2 million volt generators used to produce X rays and electron beams.

RESPIRATORY SYSTEMS: The group of organs concerned with the exchange of oxygen and carbon dioxide in organisms. In higher animals this consists successively of the air passage through the mouth, nose, and throat, the trachea, the bronchi, the bronchioles, and the alveoli of the lungs.

R-METER: (See CONDENSOR R-METER)

ROENTGENOGRAPHY: Radiography by means of X rays.

ROENTGENOLOGY: That part of radiology which pertains to X rays.

ROENTGEN RAYS: X rays.

RUTHERFORD: An obsolete unit of radioactivity equivalent to 10^6 disintegrations per second.

-S-

SARCOMA: Malignant neoplasm composed of cells imitating the appearance of the supportive and lymphatic tissues.

SCALER: An electronic device which registers current pulses received over a given time interval.

SCANNER, RECTILINEAR: A device which employs a moving collimated detector and a moving recorded to produce an image of the radionuclide distribution within an organ or gland.

SCANNING (MEDICAL): The process by which the spatial distribution of a radionuclide within an organ or gland is visualized.

SCATTERING: Change in direction of subatomic particles or photons as a result of a collision or interaction.

COHERENT SCATTERING: Scattering of photons or particles in which there are definite phase relationships between the incoming and the scattered waves. Coherence manifests itself in the interference between the waves scattered by two or more scattering centers. An example is the Bragg scattering of X rays and of neutrons by the regularly spaced atoms in a crystal, for which constructive interference occurs only at definite angles, called "Bragg angles."

COMPTON SCATTERING: The scattering of a photon by an electron. Part of the energy and momentum of the incident photon is transferred to the electron and the remaining part is carried away by the scattered photon.

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ELASTIC SCATTERING: Scattering caused by elastic collisions, and therefore conserving kinetic energy of the system. Rayleigh scattering is a form of elastic scattering.

INCOHERENT SCATTERING: Scattering of photons or particles in which the scattering elements act independently of one another, there are no definite phase relationships among the different parts of the scattered beam. The intensity of the scattered radiation at any point is obtained by adding the intensities of the scattered radiation reaching this point from the independent scattering elements.

INELASTIC SCATTERING: The type of scattering which results in the nucleus being left in an excited state and the total kinetic energy being decreased.

MULTIPLE SCATTERING: Scattering of a particle or a photon in which the final displacement is the vector sum of many - usually small - displacements.

PLURAL SCATTERING: Scattering of a particle or a photon in which the final deflection is the vector sum of a small number of displacements.

RAYLEIGH SCATTERING: The elastic scattering of a photon without loss of photonic energy. Sometimes referred to as coherent scattering.

SINGLE SCATTERING: The deflection of a particle from its original path owing to one encounter with a single scattering center in the material traversed.

SCATTERING COEFFICIENT, COMPTON: That fractional decrease in the energy of a beam of X or gamma radiation in an absorber due to the energy carried off by scattered photons in the Compton effect.

SCINTILLATION CAMERA: A device for visualizing the spatial distribution of a radionuclide within an organ or gland in the body. The gamma camera uses a stationary NaI(Tl) crystal as the detection element. Positioning signals are generated from a bank of photomultiplier tubes and applied to a cathode ray tube. Counts are integrated on film to obtain an image of the radionuclide distribution.

SCINTILLATION DETECTOR: The combination of phosphor, photomultiplier tube and associated electronic circuits for counting light emissions produced in the phosphor by ionizing radiation.

SCRAM: Emergency stopping of a nuclear reactor, usually by dropping safety rods. This may be arranged to occur automatically at a predetermined neutron flux or under other danger conditions, the reaching of which causes the monitors and associated equipment to generate a scram signal. To shut down a reactor by causing a scram.

SEALED SOURCE: A radioactive source sealed in an impervious container which has sufficient mechanical strength to prevent

GLOSSARY

contact with and dispersion of the radioactive material under the conditions of use and wear for which it was designed.

SELECTOR, PULSE HEIGHT: A circuit designed to select and pass voltage pulses in a certain range of amplitudes.

SELF ABSORPTION: Absorption of radiation emitted by radioactive atoms by the matter in which the atoms are located; in particular, the absorption of radiation within a sample being assayed.

SENSITIVE VOLUME: That portion of a counter tube or ionization chamber from which ions formed by ionizing radiation are collected and measured.

SERIES, RADIOACTIVE: A succession of nuclides, each of which transforms by radioactive disintegrations into the next until a stable nuclide results. The first member is called the "parent," the intermediate members are called "daughters," and the final stable member is called the "end product."

SHIELD: A body of material used to prevent or reduce the passage of particles or radiation. A shield may be designated according to what it is intended to absorb (as a gamma-ray shield or neutron shield), or according to the kind of protection it is intended to give (as a background, biological, or thermal shield). The shield of a nuclear reactor is a body of material surrounding the reactor to prevent the escape of neutrons and radiation into a protected area, which frequently is the entire space external to the reactor. It may be required for the safety of personnel or to reduce radiation enough to allow use of counting instruments for research or for locating contamination or airborne radioactivity.

SICKNESS, RADIATION (Radiation Therapy): A self-limited syndrome characterized by nausea, vomiting, diarrhea, and psychic depression, following exposure to appreciable doses of ionizing radiation, particularly to the abdominal region. Its mechanism is unknown and there is no satisfactory remedy. It usually appears a few hours after irradiation and may subside with a day. It may be sufficiently severe to necessitate interrupting the treatment series or to incapacitate the patient. (General): The syndrome associated with intense acute exposure to ionizing radiations.

SIGMOID CURVE: S-shaped curve, often characteristics of a dose-effect curve in radiobiological studies.

SOFTNESS: A relative specification of the quality or penetrating power of X-rays. In general, the longer the wave length the softer the radiation.

SPALLATION: A term used to denote a nuclear reaction induced by high energy bombardment and involving the ejection of more than two or three particles. (Neutrons, protons, deuterons, alpha particles, etc.).

SPECIFIC ACTIVITY: Total activity of a given nuclide per gram of a compound, element, or radioactive nuclide.

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SPECIFIC GAMMA-RAY CONSTANT: For a nuclide emitting gamma radiation, the product of exposure rate at a given distance from a point source of that nuclide and the square of that distance divided by the activity of the source, neglecting attenuation.

SPECIFIC IONIZATION: Number of ion pairs per unit length of path produced by ionizing radiations.

SPECTRUM: A visual display, a photographic record, or a plot of the distribution of the intensity of radiation of a given kind as a function of its wavelength, energy, frequency, momentum, mass, or any related quantity.

SPURIOUS COUNT: Count caused by an agency other than the radiation which it is desired to detect.

STABLE ISOTOPE: An isotope of an element which is not radioactive.

STANDARD, RADIOACTIVE: A sample of radioactive material, usually with a long half-life, in which the number and type of radioactive atoms at a definite reference time is known. It may be used as a radiation source for calibrating radiation measurement equipment.

STAY TIME: The period during which personnel are allowed to remain in a radiation and (or) contaminated area before accumulating their permissible dose.

STERILITY (BIOLOGICAL): Temporary or permanent incapability to reproduce.

STREAMING: The increase transmission of electromagnetic or particulate radiation through a medium resulting from the presence of extended voids or other regions of low attenuation. (Also called channeling effect.)

STRINGER: A long structure occupying a hole through the shield, and sometimes into the active section, of a nuclear reactor. Its removal permits access to the core for inserting experimental materials. If it is part of a large graphite reactor, for instance, part of its length may consist of graphite blocks keyed together to permit withdrawal as a unit.

STRONTIUM UNIT: A measure of the concentration of strontium-90 in food and in the body (i.e., in bone). It is measured as the ratio of strontium to calcium, with which strontium becomes mixed in soil and living tissue. One strontium unit is one picocurie (10^{-12} curie) of strontium-90 per gram of calcium.

SURVEY METER: A portable instrument with a detector, such as a Geiger counter or ionization chamber, used to detect nuclear radiation and to measure the dose rate.

SURVEY, RADIOLOGICAL: Evaluation of the radiation hazards incident to the production, use, or existence of radioactive materials or other sources of radiation under specific conditions. Such evaluation customarily includes a physical survey of the disposition materials and equipment, measurements or estimates of the levels of

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radiation that may be involved, and sufficient knowledge of processes using or affecting these materials to predict hazards resulting from expected or possible changes in materials or equipment.

SWIPE: (See WIPE SAMPLE.)

-T-

TARGET THEORY (HIT THEORY): A theory explaining some biological effects of radiation on the basis that ionization, occurring in a discrete volume (the target) within the cell, directly causes a lesion which subsequently results in a physiological response to the damage at that location. One, two, or more "hits" (ionizing events within the target) may be necessary to elicit the response.

TENTH VALUE LAYER: The thickness of a given material necessary to reduce the dose rate of an X-ray or gamma-ray beam to one-tenth its original value.

THERAPY: Medical treatment of a disease.

BRACHYTHERAPY (THERAPY AT SHORT DISTANCES): The treatment of disease with sealed radioactive sources placed near, or inserted directly into, the diseased area.

CONTACT RADIATION THERAPY: X-ray therapy with specially constructed tubes in which the target-skin distance is very short (less than 2 cm). The voltage is usually 40 to 60 kV.

RADIATION THERAPY: Treatment of disease with any type of radiation.

ROTATION THERAPY: Radiation therapy during which either the patient is rotated before the source of radiation or the source is revolved around the patient. In this way, a larger dose is built up at the center of rotation within the patient's body than on any area of the skin.

TELETHERAPY (THERAPY AT LONG DISTANCE): The treatment of disease with gamma radiation from a source located at a distance from the patient.

THERMALIZATION: Establishment of thermal equilibrium between neutrons and their surroundings. in **THERMAL NEUTRONS:** Neutrons which have come to thermal equilibrium with the atoms of a moderator, i.e., they have bounced around off moderator atoms until it is just as likely that they will gain energy as lose energy in subsequent collisions. They have a maxwellian velocity distribution and energies about the same as the thermal energies of vibration of the moderator atoms.

THRESHOLD, PHOTOELECTRIC: The quantum of energy $h\nu$ that is just enough to release an electron from a given system in the photoelectric effect. The corresponding frequency, ν , and wavelength, λ , are the threshold frequency and wavelength

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respectively. For example, in the surface photoelectric effect, the threshold $h\nu$ for a particular surface is the energy of a photon which, when incident on the surface, causes the electron to emerge with zero kinetic energy.

THRESHOLD DETECTOR: An element (or isotope) in which radioactivity is induced only by the capture of neutrons having energies in excess of a certain threshold value characteristic of the element (or isotope). Threshold detectors are used to estimate the neutrons spectrum from a nuclear (or atomic) explosion, i.e., the numbers of neutrons in various energy ranges.

TISSUE EQUIVALENT MATERIALS: Material made up of the same elements in the same proportions as they occur in a particular biological tissue. In some cases, the equivalence may be approximated with sufficient accuracy on the basis of effective atomic number.

TRACER, ISOTOPIC: The isotope or non-natural mixture of isotopes of an element which may be incorporated into a sample to permit observation of the source of that element, alone or in combination, through a chemical, biological, or physical process. The observations may be made by measurement of radioactivity or of isotopic abundance.

TRACK: Visual manifestation of the path of an ionizing particle in a chamber of photographic emulsion.

TRANSITION, ISOMETRIC: The process by which a nuclide decays to an isomeric nuclide (i.e., one of the same mass number and atomic number) of lower quantum energy. Isomeric transitions, often abbreviated I.T., proceed by gamma ray and/or internal conversion electron emission.

TRANSMUTATION: Any process in which a nuclide is transformed into a different nuclide, or more specifically, when transformed into a different element by a nuclear reaction.

-V-

VOLTAGE, OPERATING: As applied to radiation detection instruments, the voltage across the electrodes in the detecting chamber required for proper detection of an ionizing event.

VOLUME, SENSITIVE: That portion of a counter tube or ionization chamber which responds to a specific radiation.

-W-

WATER, HEAVY: Popular name for water of which the hydrogen component is deuterium.

WATT: The unit of power equal to one joule per second ($1W = 1J/s$).

GLOSSARY

WHOLE-BODY EXPOSURE: An exposure of the body to radiation where a major portion is irradiated.

WIPE-SAMPLE: A sample made for the purpose of determining the presence of removable radioactive contamination on a surface.

-X-

X RAYS: Penetrating electromagnetic radiations whose wavelengths are shorter than those of visible light.

APPENDIX

ARMED FORCES RADIATION PROTECTION PROGRAMS

1.0 INTRODUCTION

The armed forces radiation protection programs consist of plans and procedures used to protect personnel from hazardous exposure to ionizing radiation. The goal of these programs is to provide the means to maintain the health of personnel while they work in or around areas that contain radiation sources or ionizing radiation. These programs implement federal regulations related to radiation health promulgated by the Nuclear Regulatory Commission, Department of Health and Human Services, Department of Labor and the Environmental Protection Agency.

The policy of the Armed Forces is that exposure to personnel from ionizing radiation be reduced to levels as low as reasonable achievable (ALARA). Every effort is made to fulfill this objective without compromising operational efficiency and training efforts. Adequate indoctrination and training in radiological controls/radiation safety practices and protective measures are essential and required for all personnel engaged in work involving exposure to ionizing radiation.

Department of Defense organizations directly responsible for personnel working with radioactive materials or devices that produce ionizing radiation must appreciate the importance of their specific responsibilities with regard to the execution of safety and protective measures. It is mandatory that discipline be exercised to ensure that proper protective equipment and training in its use are available to and used by all involved personnel. When radiological protection measures are properly executed, the risk associated with occupational radiation exposure is small when compared to other normally encountered risks.

In the context of the Armed Forces Radiation Health Programs the term ionizing radiation is defined as any electromagnetic or particulate radiation capable of producing ions, directly or indirectly, in its passage through matter. Ionizing radiation includes the following: alpha particles, beta particles, gamma rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. The term radiation source refers to any material, equipment or device which generates or is capable of generating ionizing radiation. This includes naturally-occurring and artificially-induced radioactive material; special nuclear material; contaminated material; nuclear reactors, particle generators, and accelerators; medical, dental, and industrial radiographic and fluoroscopic equipment; and electromagnetic wave generators capable of producing ionizing radiation.

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2.0 NAVY RADIATION PROTECTION PROGRAM

The Radiation Health Protection Manual (NAVMED P-5055), a document published by the Naval Medical Command, establishes the Navy's Radiation Health Program, promulgates the radiation health requirements applicable to all radiation protection programs, and describes the radiological controls for NAVMEDCOM managed activities (Reference 1).

Radiation protection within the Navy is achieved by satisfying the requirements of NAVMED P-5055, plus the provisions of individual Radiological Controls/Radiation Safety Programs. Each of these programs, described in the following paragraphs, publish requirements and exercise oversight authority within the scope of their individual programs.

2.1 PRINCIPAL RADIOLOGICAL CONTROL/RADIATION SAFETY PROGRAMS

2.1.1 Medical Department Radiological Controls Program (NAVMEDCOM)

The Naval Medical Command is responsible for developing physical standards for all personnel of the naval establishment engaged in work involving ionizing radiation, and for applying established radiation protection standards and guidelines. It conducts basic studies on the physiological effects of ionizing radiation, and is responsible for treating radiation casualties and occupational overexposures.

NAVMEDCOM is also responsible for developing radiation health protection, personnel dosimetry, recommending training programs and qualification standards for personnel involved in radiation health protection programs, developing radiological safety programs to complement those described below, and for assistance, as required, in the radiation health aspects of nuclear accident preparedness, NBC Warfare Defense, and disaster control planning.

2.1.2 Radiological Affairs Support Program (COMNAVFACENGCOM)

The Radiological Affairs Support Program (RASP) was established in late 1971 as a result of a Chief of Naval Operations study which concluded that there was a critical need for a centrally coordinated program to provide radiological affairs support for the numerous industrial applications of ionizing radiation throughout the Navy. The area of responsibility includes Naval shore establishments and the operating fleet.

The Radiological Affairs Support Office was originally a part of the Naval Nuclear Power Unit (NNPU) located at Fort Belvoir, Virginia. In 1976 the NNPU was moved to the Construction Battalion

ARMED FORCES RADIATION PROTECTION PROGRAMS

Center, Port Hueneme, California. The Naval Nuclear Power Unit merged with the Naval Environmental Support Office in 1980 to form the Naval Energy and Environmental Support Activity (NEESA) at Port Hueneme. The NEESA provides worldwide support to Naval activities in the areas of energy, environmental protection, and occupational safety and health.

Implementation of broad-based radiation safety programs for the Naval industrial community continues to be the "raison d'être" for the Radiological Affairs Support Office. Its involvement extends to all sources of ionizing radiation and includes over 150 Navy and Marine Corps Commands. Only those sources associated with the Naval Nuclear Propulsion Program and those managed by the Naval Medical Command are excluded.

The five major areas where the Radiological Affairs Support Office provides radiation safety related services are:

Radiological Affairs Technical Assistance Visits. RASP's staff of highly trained and experienced health physicists and technicians provide extensive on-site technical services to Navy activities which possess or use any kind of ionizing radiation source. These sources are not restricted to radioisotopes and x-ray units, but include such devices as electron microscopes, electron beam welders, and particle accelerators.

Technical services include evaluating overall radiation safety programs, conducting radiation safety surveys, providing advice on disposal of radioactive materials, evaluating compliance with the Radiation Health Program managed by the Naval Medical Command, and ensuring compliance with Federal rules and regulations and conditions imposed by NRC licenses.

Nuclear Regulatory Commission Licensing Coordination. All applications for, and amendments to, NRC licenses held by Naval activities are reviewed by the Radiological Affairs Support Office before being submitted to the Nuclear Regulatory Commission. In addition, RASP personnel review proposed and final NRC rules for effects on Naval activities and provide guidance for interpretation of the regulations.

Shielding Design Review. Design review is provided for new facilities or when existing facilities are modified in ways which might affect radiation protection. Facilities reviewed include those using medical or dental x-ray machines, industrial radiography installations using large x-ray machines and accelerators, and facilities employing radionuclides such as cobalt-60 and cesium-137. The Radiological Affairs Support Office also designs radiation shielding and provides recommendations for ancillary safety devices designed to prevent inadvertent personnel exposure.

Radiological Safety Training. A number of training programs for a wide range of radiation safety applications have been

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developed. One of these programs is a 2-week Radiation Safety Officer Course developed to meet the needs for different levels of radiation safety officer training throughout the industrial, research, and training areas of the Navy.

Nuclear Weapons Radiation Safety Surveys. The Radiological Affairs Support Office assists the Naval Sea Systems Command in conducting nuclear weapons radiation surveys to verify that radiation exposures to personnel obeys the ALARA (as low as reasonably achievable) principle.

2.1.3 Nuclear Weapons Radiological Control Program (COMNAVSEASYSOM)

The Naval Sea Systems Command (SEA-644) is the technical manager for the Nuclear Weapons Radiological Control Program whose functional responsibilities lie in the following areas:

Technical Direction. Act as Technical Manager for storage, maintenance and handling of nuclear weapons. Establish Naval Nuclear Weapons Radiological Controls policy and programs. Coordinate such policies and programs with those of other government agencies and industry. Coordinate the application of commercial and regulatory whole body low-level ionizing radiation protection standards to Navy ships and activities with a nuclear weapons capability.

Operations. Issue technical guidance to Fleet Commanders, Type Commanders, Ships, and Shore activities. Publish and maintain the Nuclear Weapons Radiological Controls Manual. Obtain data from activities and issue an annual summary of radiation exposures. Provide shielding design criteria, as required, to reduce radiation exposures from Navy Nuclear Weapons in conformity with the ALARA principle. Coordinate the SEA-644 portion of the Fleet Modernization Program.

Training. Act as curriculum control authority and technical monitor for all Nuclear Weapons Radiological Controls Training. Assist Fleet Commanders in the development of course outlines and lesson plans to train ship and station Nuclear Weapons Radiological Controls Officers and Nuclear Weapons Radiological Controls Technicians.

2.1.4 Naval Nuclear Propulsion Radiological Controls Program (COMNAVSEASYSOM)

The Naval Nuclear Propulsion Program, in accordance with Presidential Executive Order 12344 of February 1, 1982, is

ARMED FORCES RADIATION PROTECTION PROGRAMS

responsible for control of radiation and radioactivity associated with Naval nuclear propulsion activities. In addition, it prescribes and enforces standards and regulations for these areas as they affect the environment and the safety and health of workers, operators, and the general public.

The interested reader is referred to the Naval Sea Systems Command for a more detailed description of this program.

2.2 RADIACS IN THE NAVY PROGRAM

The term RADIAC is an acronym for Radioactivity Detection, Indication, and Computation. It is commonly used by Army, Navy, and Air Force personnel to describe all equipments employed in the detection and measurement of radioactive sources and ionizing radiation fields. The list includes survey meters, probes, dosimeters, chargers, readers, calibrators, scalars, spectrometers, air samplers, and particle detectors.

Table 1 lists the principle Radiacs in the Navy Radiation Protection Program. Reference 2 contains shipboard allowance lists for Radiacs. A more complete list of Radiacs in the Navy Radiac Program is contained in Reference 3.

2.3 NAVY RADIAC CALIBRATION PROGRAM

The Naval Electronic Systems Command (NAVELEX) is responsible for material support functions (as described in Chapter III of Reference 4) for all Radiacs, as well as for systems integration of all Radiac systems and subsystems in the naval establishment. The NAVELEX Radiac maintenance and calibration organization is shown in Figure 1.

Reference 5 specifies responsibilities of Naval personnel and activities in the use, maintenance and calibration of Radiac equipment, and describes the requirements governing the possession and use of Radiacs containing radioactive check sources.

NAVELEX has developed a calibration and maintenance program to ensure that required numbers of calibrated and properly maintained Radiacs are available in the field. Basically, this maintenance and calibration program operates as follows: End-users (i.e., the organizations described in Section 2.1) periodically transfer the Radiac equipment on their authorized allowance list to specialized facilities. Maintenance, calibration and any necessary repair is performed on the Radiacs at these facilities and the equipment is then returned to the end-user.

There are four key elements in the Radiac Maintenance and Calibration Program, namely, the Radiac Repair Facilities (RRF), the requirement for periodic maintenance and calibration of Radiacs, the

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TABLE 1 PRINCIPAL RADIACS IN THE NAVY PROGRAM

RADIAC	OPERATING RANGE	INSTRUMENT CLASS	PRINCIPAL USE
<u>SURVEY METERS – LOW RANGE</u>			
AN/PDR-27*	0–500 mR/hr	GM Counter	β, γ Detection
AN/PDR-66*	0–1000 mR/hr	GM Counter	β, γ Detection
<u>SURVEY METERS – HIGH RANGE</u>			
AN/PDR-43*	0–500 R/hr	GM Counter	β, γ Detection
AN/PDR-45*	5–500 R/hr	GM Counter	β, γ Detection
AN/PDR-65*	0–1000 R/hr 0–10,000 R	Ionization Chamber	γ Detection
CDV-715**	0–500 R/hr	GM Counter	γ Detection
<u>ALPHA SURVEY METERS</u>			
AN/PDR-56*	0–10 ⁶ CPM (4 Ranges)	Scintillation Detector	α Detection
AN/PDR-60***	0–2X10 ⁶ CPM (6 Ranges)	Scintillation Detector	α Detection
<u>NEUTRON SURVEY METERS</u>			
AN/PDR-70*	0–2000 mrem/hr (4 Ranges)	Proportional Counter	n Detection
NP-1S****	0–2000 mrem/hr (4 Ranges)	Proportional Counter	n Detection
<u>DOSIMETERS (SELF-INDICATING)</u>			
IM-9()/PD*	0–200 mR	Quartz Fiber Ionization Chamber	γ Detection
IM-135()/PD*	0–5 R	Quartz Fiber Ionization Chamber	γ Detection
IM-143()/PD*	0–600 R	Quartz Fiber Ionization Chamber	γ Detection

ARMED FORCES RADIATION PROTECTION PROGRAMS

TABLE 1 PRINCIPAL RADIACS IN THE NAVY PROGRAM (CONT.)

RADIAC	OPERATING RANGE	INSTRUMENT CLASS	PRINCIPAL USE
<u>DOSIMETERS (NON SELF-INDICATING)</u>			
DT-60/PD ^{(1)*}	0-600 R	RPL	γ Detection
DT-518/PD*	10-50,000 rem 1-10,000 R	Neutron Activation	n Detection
DT-526/PD ^{(2)*}	5 mR-5000 R 10-50,000 R	TLD	γ Detection
<u>CALIBRATORS</u>			
AN/UDM-1A*	—	Cs-137 (120 Ci)	γ Source
AN/UDM-7A*	—	Pu-239 (51 μ Ci)	α Source
AN/UDM-5*	—	PuBe (5 Ci)	n Source
AN/UDM-10*	—	AmBe (100 μ Ci)	n Source
TS-1189()/PD*	—	Cs-137 (700 μ Ci)	γ Source
TS-1216()/UD*	—	Cs-137 (30 Ci)	γ Source
<u>AIR SAMPLERS / PARTICLE DETECTORS</u>			
HD-251/UD*	—	—	Air Sampler
IM-192/WDQ*	10^{-9} - 10^{-7} μ Ci/cm ³	Scintillation Detector	Particle Detector

* STANDARD — Most advanced and satisfactory equipments approved for service use.

** SUBSTITUTE STANDARD — Approved for service use, but not as satisfactory as standard equipments.

*** OBSOLESCECENT — Equipments which no longer have satisfactory military characteristics, but which must be continued in service pending availability of improved replacements.

**** LIMITED STANDARDS — Equipments approved for service use which do not have as satisfactory military characteristics as standard or substitute equipments, but are useable substitutes.

(1) CP-95/PD Reader, (2) CP-1112/PD Reader.

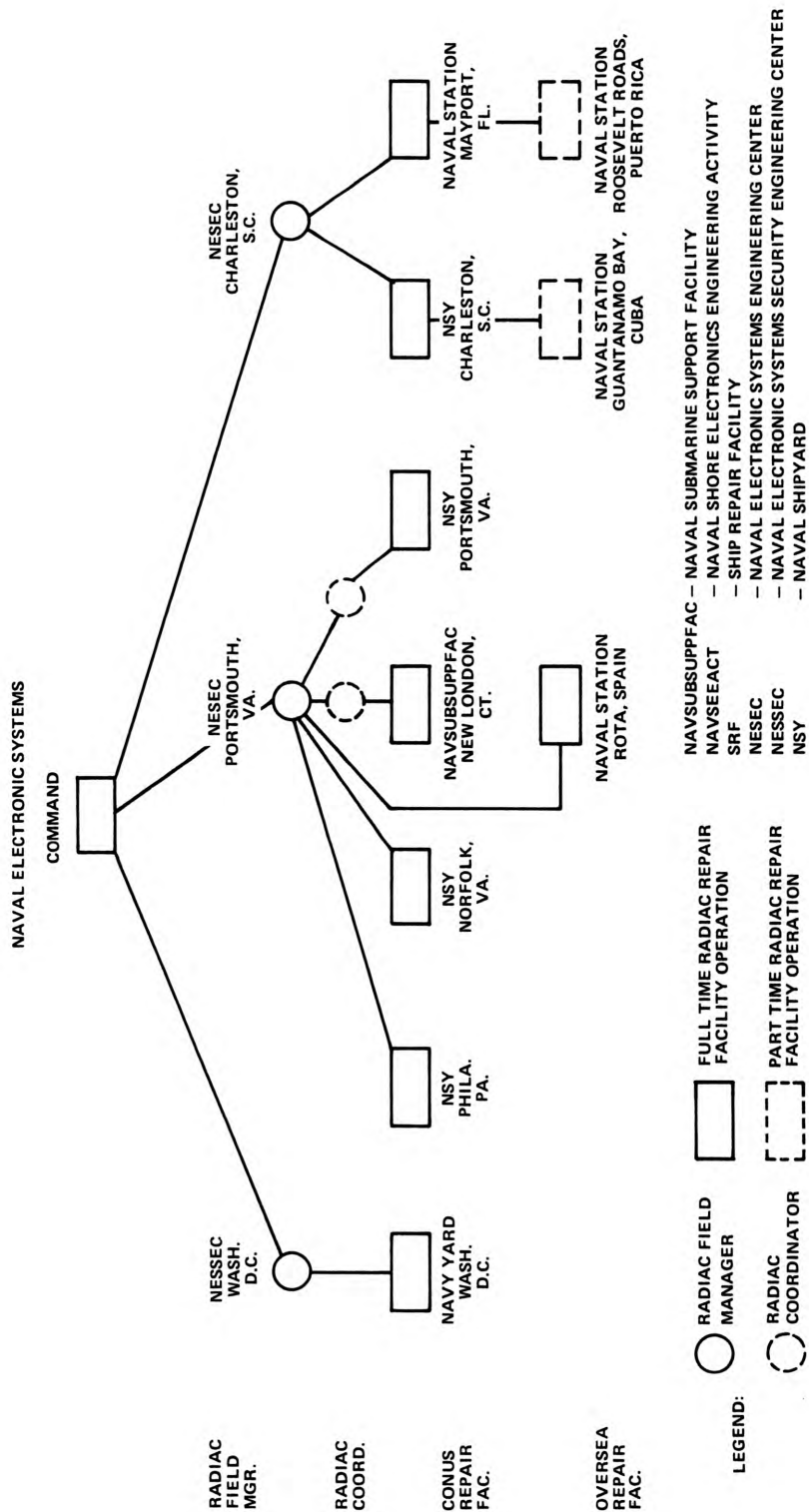


FIGURE 1(A) NAVELEX RADIAC MAINTENANCE AND CALIBRATION ORGANIZATION (EAST COAST AND ATLANTIC).

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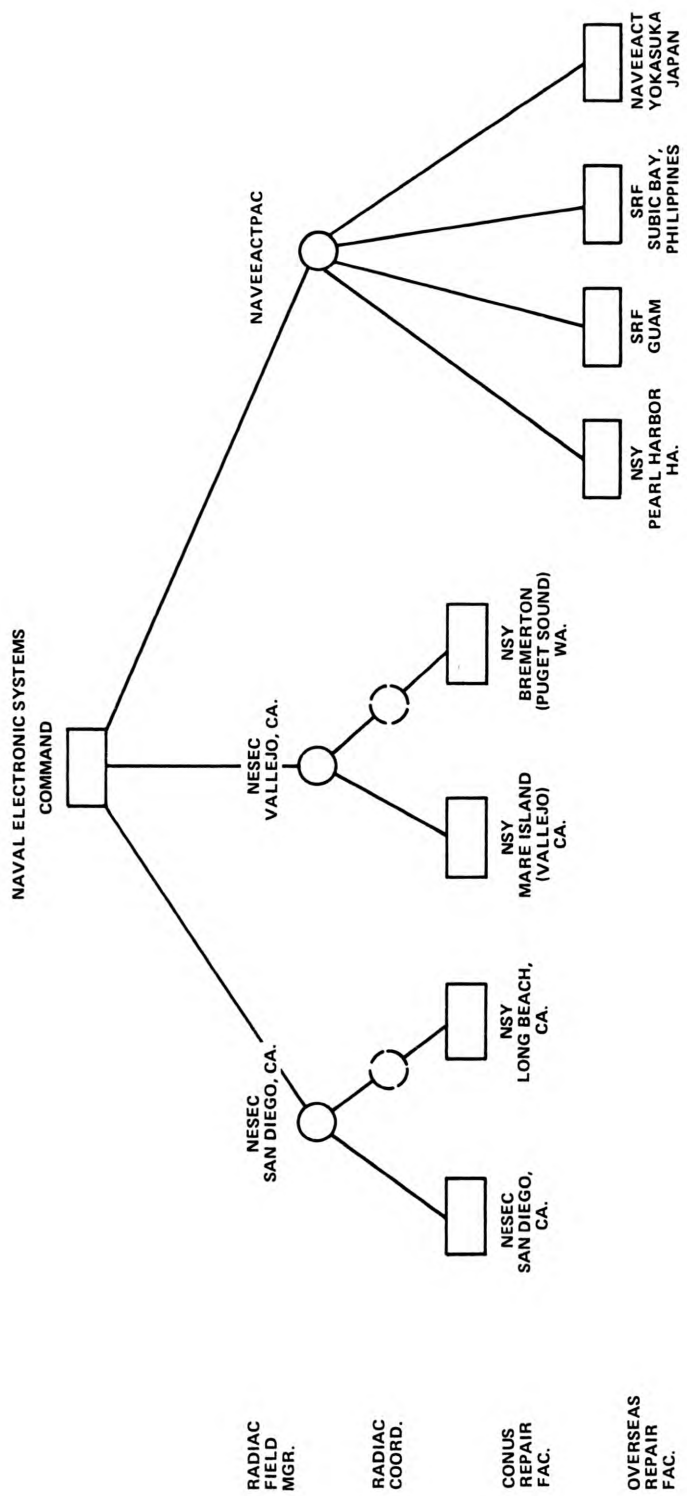


FIGURE 1(B) NAVELEX RADIAC MAINTENANCE AND CALIBRATION ORGANIZATION (WEST COAST AND PACIFIC).

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MEASURE program, and the Radiac Field Manager/Radiac Coordinator.

2.3.1 Radiac Repair Facility

The calibration of many Radiacs involves the use of relatively large radioactive sources. Consequently, specially equipped facilities for the maintenance and calibration of Radiacs have been established. These facilities, called Radiac Repair Facilities (RRF), are operated by personnel trained in all aspects of Radiac maintenance and calibration, and are designed to allow the safe use of multicurie calibration sources. RRF's are located on all tenders with nuclear support facilities and at the shore activities listed in Table 2.

2.3.2 Periodic Maintenance and Calibration

The NAVELEX Radiac calibration procedures manual (Reference 6) codifies procedures for proper calibration of almost all Radiac equipments. Calibration is accomplished in most cases by periodic transfer of the Radiacs from the end-user to a RRF. The calibration intervals for Radiac equipment are given in Enclosure (5) of Reference 5. Not all Radiacs are transferred to a RRF for calibration. The Radiacs listed in Enclosure (6) of Reference 5 are calibrated by the end-user or are calibrated by NAVELEX personnel at the end-user's activity.

2.3.3 The MEASURE Program

In order to facilitate efficient operation of the Navy calibration program the Naval Material Command has recently implemented a Metrology Automated System for Uniform Recall and Reporting (MEASURE) Program.

The MEASURE program (as applied to Radiac maintenance and calibration) can be expected to produce the following benefits: 1) enable end-users to be notified by monthly MEASURE reports when specific Radiacs are due for calibration, 2) generate relevant data which the Metrology Engineering Center can use to formulate recommendations to NAVELEX concerning optimal calibration cycles for Radiac, and 3) provide accurate information on locations of all equipments resulting in more efficient management of Radiac inventories and allowances.

Information on individual Radiac equipments is recorded by end-user activities and RRF's on forms called MEASURE Meter Cards. Details concerning the operation of the MEASURE program can be found in Reference 7 and 8.

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**TABLE 2 LOCATION OF RADIAC FIELD MANAGERS AND RADIAC COORDINATORS,
LIST OF NAVELEX APPROVED RADIAC REPAIR FACILITIES AND AREAS
OF RESPONSIBILITY.**

LOCATION OF RADIAC FIELD MANAGER OR RADIAC COORDINATOR	NAVELEX APPROVED RADIAC REPAIR FACILITY	ASSIGNED MAINTENANCE AND CALIBRATION RESPONSIBILITY
Portsmouth Naval Shipyard Portsmouth, NH 03801 (Radiac Coordinator)	Portsmouth Naval Shipyard, NH	Maine, Vermont, NPTU West Milton, NY, Massachusetts (North of Boston), New Hampshire, NPTU Windsor, CT
Naval Submarine Support Facility, New London Groton, CT 06340 (Radiac Coordinator)	Naval Submarine Support Facility, New London, CT	Connecticut, Rhode Island, Massachusetts (South of Boston), Boston
Naval Electronic Systems Engineering Center, Portsmouth P. O. Box 55 Portsmouth, VA 23705 (Radiac Field Manager)	Naval Station Rota, Spain	Naval Forces Europe, No. Africa
	Philadelphia Naval Shipyard, PA	New York, New Jersey, Penn., Ohio, Indiana, Michigan
	Norfolk Naval Shipyard, VA	Southern VA, North Car., Kentucky, Bermuda, Iceland, Newfoundland
Naval Electronic Systems Security Engineering Center Naval Security Station 3801 Nebraska Ave., NW Washington, DC 20309 (Radiac Field Manager)	Washington Navy Yard, DC	Delaware, Maryland, No. VA., District of Columbia, West VA.
Naval Electronic Systems Engineering Center, Charleston 4600 Goer Road North Charleston, SC 29406 (Radiac Field Manager)	Naval Station Mayport, FL	Alabama, Mississippi, Missouri, Oklahoma, Georgia, Minnesota, Wisconsin, Illinois, Kansas, Ark., Louisiana, Texas, Tennessee, FLA, Iowa, So. Car., (less Charleston area), Caribbean Area, Puerto Rico, Panama Canal
	Charleston Naval Shipyard, SC	Charleston area only
Naval Electronic Systems Engineering Center, San Diego P. O. Box 80337 San Diego, CA 92138 (Radiac Field Manager)	NAVELEXSYSENGCEN San Diego, CA	Southern California (less Long Beach Naval Shipyard), Arizona, New Mexico

ARMED FORCES RADIATION PROTECTION PROGRAMS

**TABLE 2 LOCATION OF RADIAC FIELD MANAGERS AND RADIAC COORDINATORS,
LIST OF NAVELEX APPROVED RADIAC REPAIR FACILITIES AND AREAS
OF RESPONSIBILITY. (CONT.)**

Long Beach Naval Shipyard Long Beach, CA 90822 (Radiac Coordinator)	Long Beach Naval Shipyard, CA	Long Beach Naval Shipyard area only
Naval Electronic Systems Engineering Center, Vallejo Vallejo, CA 94592 (Radiac Field Manager)	Mare Island Naval Shipyard Vallejo, CA Naval Reactors Facility Idaho Falls (NRF)	Northern California, Nevada, Colorado, Utah Prototype sites at NRF only
Puget Sound Naval Shipyard Bremerton, WA 98314 (Radiac Coordinator)	Puget Sound Naval Shipyard Bremerton, WA	Alaska, Idaho (less Idaho Falls NRF), Washington, Oregon, Montana, Wyoming, No. Dak.
Naval Shore Electronics Engineering Activity, Pacific P. O. Box 130 Pearl Harbor, HI 96860 (Radiac Field Manager)	Pearl Harbor Naval Shipyard, HI	Hawaii, Australia, New Zealand, Antarctica
Naval Shore Electronics Engineering Activity Yokosuka, Japan (Radiac Coordinator)	Naval Shore Electronics Engineering Activity Yokosuka, Japan	Japan, Okinawa, Korea
Naval Ship Repair Facility Guam (Radiac Coordinator)	Naval Ship Repair Facility Guam	Naval Forces, Marianas
Naval Ship Repair Facility Subic Bay (Radiac Coordinator)	Naval Ship Repair Facility Subic Bay	Naval Forces, Philippines, Southeast Asia, Diego Garcia
N/A	All tenders with nuclear support facilities	Assigned by applicable type commander

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2.3.4 Radiac Field Manager/Radiac Coordinator

The Radiac Field Manager (or Radiac Coordinator) resolves questions that the end-users and/or the RRF's may have about the use, maintenance, and calibration of all Radiac equipment. In addition, as a NAVELEX representative, the Radiac Field Manager (or Radiac Coordinator) implements policy concerning Radiac use, maintenance, and calibration. Specific duties of the Radiac Field Manager and Radiac Coordinator are described in Enclosure (7) of Reference 5.

3.0 ARMY RADIATION PROTECTION PROGRAM

3.1 RADIOLOGICAL CONTROL/RADIATION SAFETY PROGRAM

3.1.1 Regulations

The Army's radiological control and radiation safety program is defined by various Army Regulations (AR's). Table 3 lists the major Department of Army Regulations related to radiation protection. Additional AR's and subordinate command regulations are in force, as well.

3.1.2 Licenses, Authorizations, and Permits

NCR licenses are required for Army activities withing the United States, its territories and possessions to produce, transfer, receive, own, possess, use, and dispose of byproducts, source and special nuclear material in excess of license-exempt quantities in 10 CFR 30-34, 40 and 70. Generally, Army NRC license applicants will follow one of three routes to the NRC. Medical type applications will be routed through the DA Surgeon General for review and approval prior to being forwarded to the NRC. Applications involving civil works are routed through the DA Corps of Engineers. Other types are processed by the Army Development and Readiness Command (DARCOM) Safety Office.

Department of Army Radiation authorizations are generally required for Army activities having radioactive material, but not requiring an NRC license. Department of Army Radiation Permits are required for non-Army agencies, including civilian contractors, who use, store, possess, or dispose of radiation sources on Army installations. Licenses, authorizations and permits are established and issued in accordance with AR 385-11.

ARMED FORCES RADIATION PROTECTION PROGRAMS

TABLE 3 MAJOR DEPARTMENT OF ARMY RADIATION PROTECTION REGULATIONS.

<u>AREA</u>	<u>REGULATION NO.</u>	<u>TITLE</u>	<u>SUBJECTS</u>
Inspections And Investigations	AR 20-1	Inspector General Activities and Procedures	Nuclear Surety Inspectors General Qualifications And Programs
Medical Services	AR 40-5	Health & Environment	Radiological Hygiene
Medical Services	AR 40-14	Control & Recording Procedures For Exposure To Ionizing Radiation and Radioactive Materials	Defines Exposure Standards, Utilization Of Dosimetry Devices, Recording Dosimetry Records, Control Of Exposure Levels, Etc.
Medical Services	AR 40-37	Licensing & Control of Radioactive Materials For Medical Purposes	(Self-explanatory)
Environment	AR 200-1	Environmental Protection & Enhancement	(Self-explanatory)
Environment	AR 200-2	Environmental Effects of Army Actions	(Self-explanatory)
Safety	AR 385-11	Ionizing Radiation Protection	Licensing & Control Of Radiation Sources, Controlled Radioactive Items Of Supply, Transportation, And Disposal Of Unwanted Radioactive Material.
Safety	AR 385-80	Nuclear Reactor Health and Safety	(Self-explanatory)
Logistics	AR 700-64	Radioactive Commodities In The DOD Supply System	Control Of DOD Radioactive Commodities In All Phases Of Their Life Cycle.

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3.1.3 Documentation

DA publications document various aspects of the radiation safety program. Army Radiac meters are furnished with technical manuals (TMs) to instruct in the instrument's proper use. TMs are also furnished with the calibration sources to ensure their safe and proper use in calibrating Radiac meters. Technical bulletins (TBs) are also provided to describe the proper calibration procedures for RADIAC meters, for defining the proper handling of radioactive sources and for instruction in the proper handling and disposal of (unwanted) radioactive material.

3.1.4 Radiological Protection Officers (RPO)

The Army utilizes physicians, health physicists and other personnel assigned as RPO's to administer its radiation safety program. RPO's may be civilian or military, officers or enlisted men. RPO's are trained in the basics of radiation protection, radiation physics, the measurement of radiation, and the laws and regulations pertaining to radiation safety. Some of the responsibilities of a health physicist or RPO are verifying that work conditions comply with regulations and directives; instructing personnel in safe working practices, emergency procedures, harmful effects of radiation, etc., and evaluating and correcting radiological hazards. The RPO also performs periodic surveys and wipes of areas with radioactive sources and other potential radiation hazards. Wipe results of items managed by commodity commands are set to the commands where records of source location and wipe history are kept.

Local ionizing radiation control committees (IRCC) also exist to provide assurance that radiation safety doesn't break down. The IRCC reviews proposals to use or procure radioactive items (through review of license applications and SOP's), considers ways to improve existing safety programs, and studies reports of radiological incidents.

Training for RPO's and radiation workers is available from the US Army Chemical School at Fort McClellan, AL. Training is also available through MERADCOM at Fort Belvoir, VA and through local informal OJT programs. University short courses in radiation protection are also utilized.

3.1.5 Dosimetry

Service for technical radiation exposure control is provided to all Army radiation workers by the Army Ionizing Radiation Dosimetry Center (AIRDC), Lexington-Blue Grass Depot, Kentucky. This service consists of film and TLD badges, which are issued to

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radiation workers for a wearing period of 4 or 5 weeks. Film usage is being phased out and will eventually be replaced entirely by TLDs.

Exposure of individuals are determined by returning the film or TLD to AIRDC for reading. All readings are permanently recorded in a file for each individual which constitutes their official Army radiation exposure record. Self-reading pocket ion chambers or other portable dosimeters are also available for individual worker use.

3.1.6 Inspectors

Army activities with NRC licensed radioactive material are subject to being inspected by the NRC, the Inspector General (IG), the Army Environmental Hygiene Agency (AEHA), the DARCOM Field Safety Agency (if a part of DARCOM), and by organizational and local safety inspectors. Additionally, the Nuclear Surety Agency inspects organizations which possess or support nuclear weapons. Army organizations with DA authorizations or permits to possess radioactive material, as well as activities having custody of radioactive items licensed by the NRC to an Army commodity manager (commodity command), are also subject to safety inspections. The purpose of these inspections is to assure that danger to personnel and property is minimized through enforcement of laws, regulations, policies, and SOP's.

3.2 RADIACS IN THE ARMY PROGRAM

The term RADIAC is a acronym for Radioactivity Detection, Indication, and Computation. It is commonly used by Army, Navy, and Air Force personnel to describe all equipments employed in the detection and measurement of radioactive sources and ionizing radiation fields. Table 4a lists the Radiacs in the Field Army, i.e., standardized, ruggedized instruments for field use. Commercial Radiac instruments are also used by the "industrial" Army for support of research, testing and evaluation, health physics and metrology laboratories. Army commercial Radiacs are not listed because they are subject to frequent change. Table 4b lists the Field Army Radiac Calibration Standards. These standards can be used to calibrate all of the Radiacs in Table 4a as well as a number of commercial Radiac meters.

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TABLE 4a FIELD ARMY RADIACS

<u>RADIAC TMDE</u>	<u>NOUN</u>	<u>DESCRIPTION</u>	<u>CALIBRATION STANDARD</u>
AN/PDR-27()	Beta, gamma survey meter	Survey meter with thin mica end window GM tube for 0 - 0.5 & 0 - 5.0 mR/hr ranges, and second GM tube for 0 - 50 & 0 - 500 mR/hr ranges.	AN/UDM-2 or gamma calibration source
AN/PDR-54	Alpha survey meter	Gas flow proportional alpha counter with ranges of 0 - 10 ³ , 0 - 10 ⁴ , and 0 - 10 ⁵ CPM.	AN/UDM-6 or AN/UDM-7()
AN/PDR-56	Alpha survey meter	Scintillation alpha counter with ranges of 0 - 10 ³ , 0 - 10 ⁴ , 0 - 10 ⁵ , 0 - 10 ⁶ CPM.	AN/UDM-7()
AN/PDR-60	Alpha survey meter	Scintillation alpha counter with ranges of 0 - 2 x 10 ³ , 0 - 2 x 10 ⁴ , 0 - 2 x 10 ⁵ , 0 - 2 x 10 ⁶ CPM.	AN/UDM-6 or AN/UDM-7()
IM-174()/PD	Gamma survey meter	Ion chamber survey meter with range of 1 to 500 R/hr.	AN/UDM-2 or gamma calibration source
IM-9()	Pocket dosimeter	0 - 200 mR gamma quartz fiber, self-indicating pocket dosimeter.	"
IM-93	Pocket dosimeter	0 - 600 R gamma quartz fiber, self-indicating pocket dosimeter.	"
IM-147	Pocket dosimeter	0 - 50 R gamma quartz fiber, self-indicating pocket dosimeter.	"
IM-185	Pocket dosimeter	0 - 600 R gamma and neutron quartz fiber, self-indicating pocket dosimeter.	"

TABLE 4b FIELD ARMY RADIAC CALIBRATION STANDARDS.

<u>CALIBRATION STANDARD</u>	<u>NOUN</u>	<u>DESCRIPTION</u>	<u>ITEMS SUPPORTED</u>
AN/UDM-1	Radiac Calibrator	Non-portable Gamma Source; 9 Ci Of Co-60 In Shielded Housing, With Source Housing Stand, Exposure Control Rods, Exposure Table On Carriage Assembly, Carriage Tracks & Telescope.	Large Number of Gamma Detectors (Survey Meters, Pocket Dosimeters And Radiation Chambers).
AN/UDM-1A	Radiac Calibrator	Non-portable Gamma Source; 120 Ci Of Cs-137 In Shielded Housing With Source Housing Stand, Exposure Control Rods, Exposure Table On Carriage Assembly, Carriage Tracks And Telescope.	"
AN/UDM-2	Calibrator Set, Radiac	Portable Beta Source; Total Activity Of 0.2 Ci Of Sr-Y-90, Packaged In Two Parts - A Radiac-Meter Calibrator (TS-3494/UDM-2) And Dosimeter Calibrator (TS-3495/UDM-2).	IM-174()/PD Radiacmeter AN/PDR-27() Radiac Set IM-9()/PD Dosimeter IM-93/UD Dosimeter IM-147/PD Dosimeter CD V-138 Dosimeter CD V-742 Dosimeter
AN/UDM-6	Radiac Calibrator	Four Portable Alpha Sources; Total Activity 1.4 μ Ci Of Pu-239. Sources Are Fitted In Metal "Shoes" For Mating With Alpha Radiacmeter Probes.	AN/PDR-54 AN/PDR-60
AN/UDM-7()	Alpha Radiac Calibrator	Two Portable Alpha Sources; Total Activity 10 ⁷ DPM Of Pu-239. Set Includes Attenuators, And Adaptors For Several Alpha Radiacmeter Probes.	AN/PDR-54 AN/PDR-56 AN/PDR-60
TS-1230A	Radiac Calibrator	Single Portable Alpha Source; Activity 4 μ Ci Of Pu-239. Set Includes Two Attenuators.	AN/PDR-53

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3.3 ARMY RADIAC CALIBRATION PROGRAM

3.3.1 Chain of Measurements

The measurement chain starts with international standards and works downward to lower level standards (see Figure 2). Comparisons of national radiation standards of various countries are made on an international level. Participating international organizations have included the International Commission on Radiation Units and Measurements (ICRU), the International Atomic Energy Agency (IAEA), and the International Bureau of Weights and Measures (BIPM). The BIPM is now the primary organization for comparison of national standards of radioactivity. The United States measurement standards for ionizing radiation are maintained by the National Bureau of Standards (NBS) at Gaithersburg, MD. The Radiation Physics Division at NBS provides national primary level calibration service to the Army's Primary Nucleonics Laboratories. This service is in the areas of alpha, gamma, x-ray, and neutron radiations.

The Army maintains two primary nucleonic calibration laboratories. These are located at the Lexington-Blue Grass Army Depot Activity (LBDA), Lexington, KY and at the Sacramento Army Depot (SAAD), Sacramento, CA. The highest level of radiation standards within the Army are maintained at these laboratories. These facilities support lower level nucleonic calibration facilities such as secondary reference level laboratories, and internal ionizing radiation calibration operations. See Table 5 for a list of primary nucleonic laboratory calibration capabilities. Beneath the primary level, the Army maintains secondary reference nucleonic calibration laboratories at Pirmasens, Germany and at Camp Carroll, Korea.

3.3.2 Operational Overview of US Army Ionizing Radiation Calibration System

The Army's ionizing radiation calibration program is a part of the Army's Test, Measurement and Diagnostic Equipment (TMDE) system. It is directed by the US Army TMDE Support Group (USATSG), Redstone Arsenal, AL. USATSG maintains four Army Calibration and Repair Centers (ACRC's) equipped as nucleonics laboratories. These ACRC's are at Lexington, Sacramento, Pirmasens, Germany and Camp Carroll, Korea. As their name implies, these centers provide repair and calibration of Radiac devices in their support area.

The Primary Nucleonic Laboratories at ACRC-Lexington and ACRC-Sacramento provide the support for Secondary Reference Laboratories, internal Army calibration facilities, Army hospitals and other military services. Although they are primary level calibration laboratories, they also provide secondary reference level calibration services; plus calibration of Radiac survey meters and dosimeters (except film and TLD) for customers in their geographical

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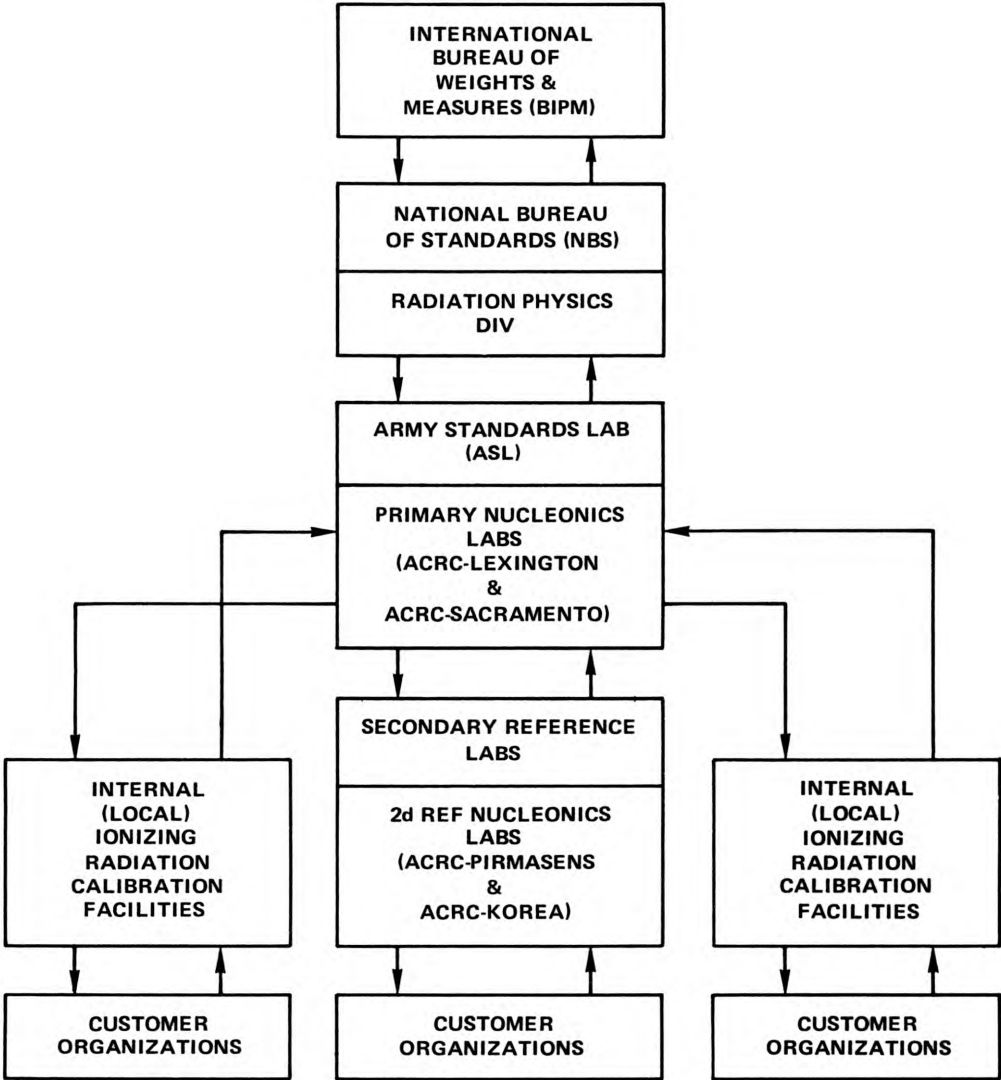


FIGURE 2 ARMY RADIAC MAINTENANCE AND CALIBRATION ORGANIZATION.

TABLE 5 US ARMY PRIMARY NUCLEONIC LABORATORIES' CAPABILITIES.

<u>PARAMETERS CALIBRATED</u>	<u>ENERGY RANGES CALIBRATED</u>	<u>EXPOSURE RATE LIMITS (METHOD OF CALIBRATION)</u>	<u>UNCERTAINTY IN CALIBRATION</u>
Alpha sources	0.4 MeV & Greater	(0 - ∞) CPM (ion chamber) (0 - 3.5 x 10 ⁶) CPM (gas proportional chamber)	± 2½% for ion chamber ± (1 1/2 - 2)% for gas proportional chamber
Beta sources	0.04 MeV (average) & Greater	(0 - 3.5 x 10 ⁶) CPM (gas proportional chamber)	± (1 1/2 - 2)%
X-ray detectors	(5 - 22) KeV	(2 - 2 x 10 ³) R/hr (free air standard)	± 2.5%
X-ray detectors	(30 - 400) KeV	4 mR/hr - 1 x 10 ³ R/hr (free air standard)	± 2%
Gamma detectors or sources	(0.05 - 1.3) MeV	.4 mR/hr - 1 x 10 ³ R/hr (Shonka chamber standard)	± 3%
Thermal neutron detectors	Thermal neutrons	(1.3 x 10 ⁴) T.N./cm ² /sec (Am-Be standard)	± 10%
Fast neutron detectors	(.5 - 14) MeV	(0.1 - 2 x 10 ⁴) N/cm ² /sec (long counter, solid state detector, MnS Bath)	± 8%
Fast neutron detectors	5.1 MeV	(5 - 2.5 x 10 ³) F.N./cm ² /sec (Pu-Be source)	± 5%

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service area. ACRC-Lexington services customers in Europe and the continental US from the states immediately west of the Mississippi river and eastward. ACRC-Sacramento services the remaining continental United States, Hawaii, Alaska, Korea, and the Pacific.

The Secondary Reference Nucleonics Laboratories at ACRC-Pirmasens and ACRC-Korea provide calibration support for Radiac standard sources (AD/UDM-2, AN/UDM-6, and AN/UDM-7) and for Radiac survey meters and dosimeters in their respective areas. Several Army installations maintain facilities for calibration of ionizing radiation instrumentation that are separate from the aforementioned four ACRCs. Such facilities are referred to as internal nucleonic calibration laboratories. Size and capabilities of these facilities vary depending upon the local calibration requirements. Aberdeen Proving Ground, MD and White Sands Missile Range, NM, for examples, possess such laboratories

Radiac devices used within the Army consist of standardized Radiacs and commercial Radiacs. The standardized Radiacs are ruggedized instruments, designed to operate over wide ranges of temperature, humidity and mechanical shock (see Tables 4a and 4b). These devices are developed by the US Army Electronics Research and Development Command, Fort Monmouth, NJ, per requirements specified by the US Army Training and Doctrine Command. Subsequent Radiac life cycle management (documentation, safety, deployment, etc.) is the responsibility of the US Army Communications-Electronics Command, Fort Monmouth, NJ. Many commercial Radiac meters are used by (non-field) Army organizations such as research laboratories, health physics organizations, hospitals and safety offices. Calibration and repair of both types of Radiac instruments is provided by the Army TMDE Support System. A detailed description of the Army Calibration Program of Radiac Survey Meters can be found in TB 9-6665-285-15.

4.0 AIR FORCE RADIATION PROTECTION PROGRAM

The reader is referred to AFOSH Standard 161-15 (in preparation) for a description of the Air Force radiation protection program.

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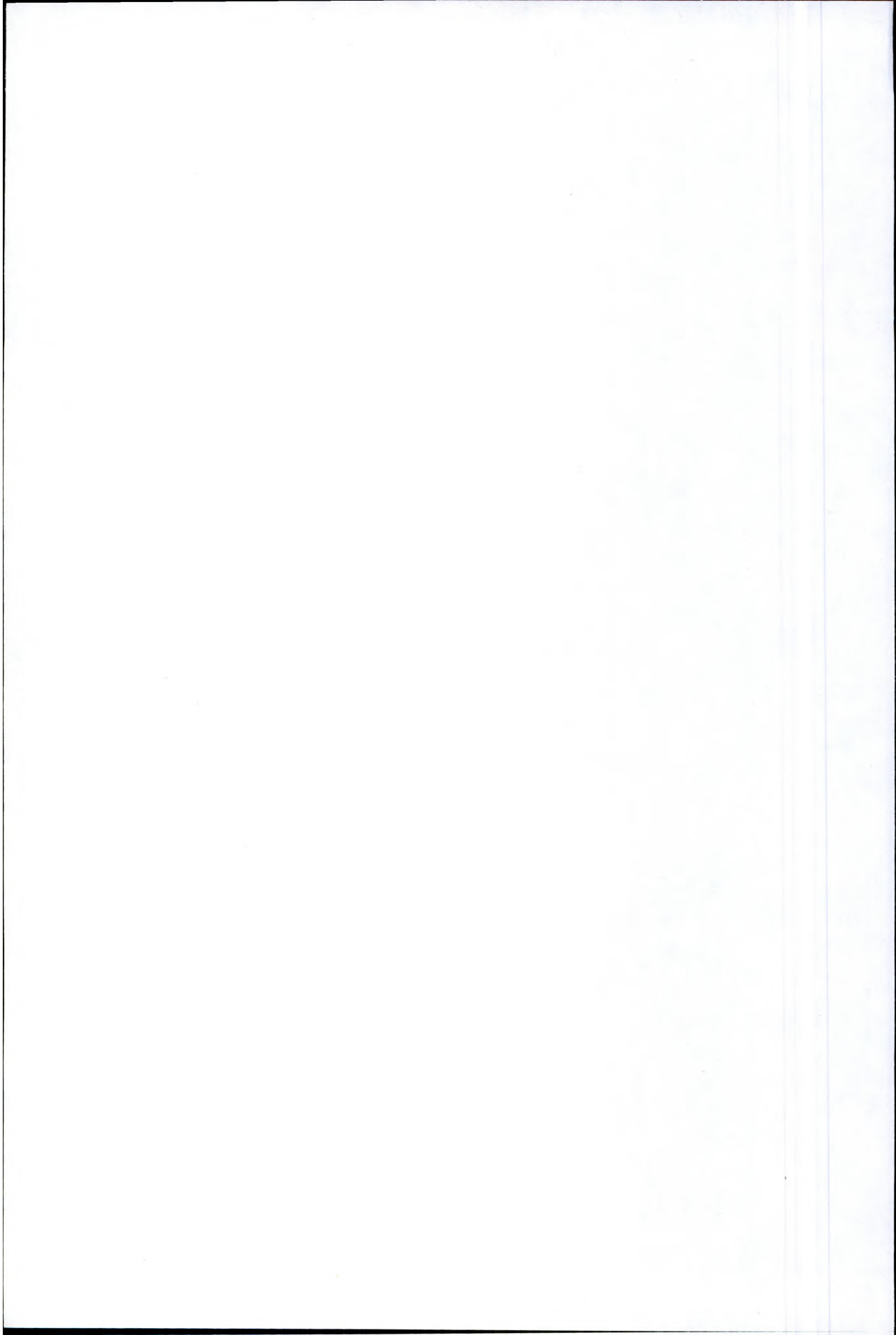
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